ENCLOSURE 2 TO AEP-NRC-2014-59

I&M CAP Document AR 2010-1804-10, Root Cause Evaluation Attachment, “Rx Vessel Core Support Lug Bolting Anomalies”

PREVENTION
DETECTION
CORRECTION

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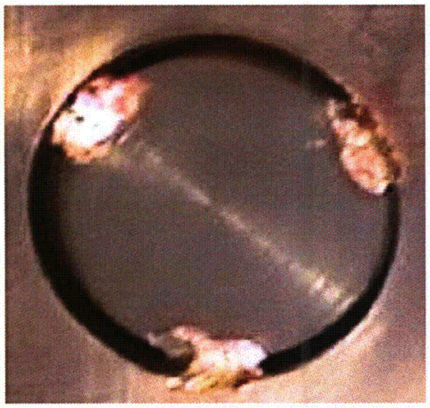

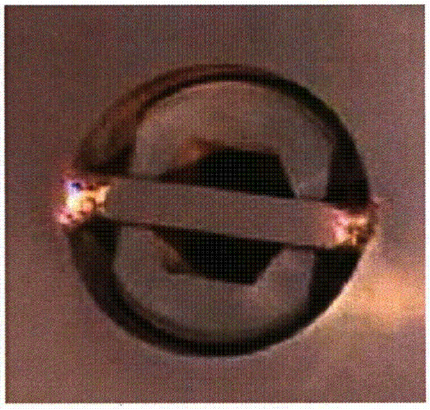

Root Cause Evaluation of Unit 1 Rx Vessel Core Support Lug Bolting Anomalies

(AR 2010-1804-10)
March 21, 2010

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1.0 EXECUTIVE SUMMARY

1.1 PROBLEM STATEMENT

Lower Radial Support System (LRSS) clevis insert bolts and dowel pin failed unexpectedly resulting in the organization addressing emergent concerns of structural margin and loose parts in the reactor coolant system (RCS).

1.2 EVENT DESCRIPTION

During the 10-year in-service inspection (ISI) of the Donald C. Cook Nuclear Plant Unit 1 (D. C. Cook Unit 1) reactor vessel in March 2010, an anomaly was noted concerning the LRSS clevis inserts. Seven (of 48) bolt locations had wear observed on the lock bar, indicating that the cap screw head had detached from the shank and that flow was causing the head to vibrate against the lock bar. One (of 12) dowel pins had broken tack welds and it was rotated and displaced into the clevis insert.

1.3 ROOT CAUSE(S) & CORRECTIVE ACTION(S) TO PRECLUDE REPETITION

Primary water stress corrosion cracking (PWSCC) of the clevis insert bolts due to the use of Alloy X-750 with a susceptible heat treatment.

1.3.1 CORRECTIVE ACTIONS TO PRECLUDE REPETITION

Developed and implemented a minimum bolting pattern under EC-51640, "RX Vessel Lower Radial Support System (LRSS) Clevis Replacement Bolting for Unit 1," which incorporated an improved heat treatment and an under head radius that reduced peak stresses.

WO# 55399712	Owner: TPG	Completion Date: 9/28/2013

Basis:

The issues reviewed in this evaluation showed the failure of clevis insert bolts does not have a safety or operational impact on the plant. The observed condition was repaired with a qualified minimum bolt pattern. Any additional inspections, repairs, pre-emptive replacements, or other actions are performed per management discretion.

A meeting was held on January 20, 2014 to discuss corrective actions. The attendees came to agreement on items entered into the CAP system as a result of this evaluation, including absence of corrective actions driving commercial issues. The following is a list of attendees:

- Engineering Vice President
- Site Vice President
- Performance Improvement Manager

- Regulatory Affairs Manager
- Design Engineering Director
- Mech. and Struct. Design Engineering Manager
- Design Engineering Mechanical Supervisor
- Reactor Vessel Internals Engineer
- Plant Engineering Director
- Operations Refueling

1.4 EXTENT OF CAUSE & CORRECTIVE ACTIONS

The cause of the observed failures was PWSCC of the clevis insert bolts due to the use of Alloy X-750 with a susceptible heat treatment. The extent of cause evaluation reviewed components that met all of the following criteria as these could be subjected to a similar cause and result in loose parts in the RCS.

- Alloy X-750 or other nickel based alloys
- Inside the reactor vessel
- Exposed to primary water

The extent of cause also focused on components that meet the criteria above that are not covered under a specific program, or where programmatic elements may overlap as these could contribute to a similar consequence of having to address emergent concerns upon failure. The identified components that warranted corrective actions are listed below. A full evaluation of the extent of cause is provided in Attachment 5.2.

- Clevis Insert Bolts
- Clevis Dowel Pins
- LRSS Lug Weld
- BMI Nozzles and Welds

Additionally, a rigorous component review of the reactor vessel internals is being performed to support the use of MRP-227-A in the Reactor Vessel Internals Aging Management Program. CA# 2010-1804-31 has been created to review the results of the component evaluation and create actions as necessary based on the causes of the LRSS bolt failure.

1.4.1 EXTENT OF CAUSE CORRECTIVE ACTIONS

Determine if the Alloy 600 LRSS clevis insert dowel pins should be added to the Alloy 600 program/procedure since they are currently not identified in EHI-5070-ALLOY600, "Alloy 600 Material Management Program".

CA# 2010-1804-37	Owner: ENU	Due Date: 4/14/2014

Evaluate if potential mitigating actions exist, such as peening, to prevent PWSCC of the LRSS lug weld and generate corrective actions to implement mitigating techniques if warranted.

CA# 2010-1804-29	Owner: ENU	Due Date: 4/14/2014

Evaluate if potential mitigating actions exist, such as peening, to prevent PWSCC of the BMIs and generate corrective actions to implement mitigating techniques if warranted.

CA# 2010-1804-30	Owner: ENU	Due Date: 4/14/2014

Review the results from contract 1500016-156 and create actions to address PWSCC aging effects for any newly identified nickel based alloys based on the root cause of the LRSS bolt failures as necessary.

CA# 2010-1804-31	Owner: DEM	Due Date: 6/18/2014

1.5 CONTRIBUTING CAUSE(S) AND CORRECTIVE ACTION(S)

Localized peak stresses at the LRSS clevis insert bolt head to shank radius due to inadequate bolt design dimensions.

1.5.1 CORRECTIVE ACTIONS

See corrective actions to preclude repetition.

1.6 OTHER CORRECTIVE ACTION(S)

1.6.1 INTERIM CORRECTIVE ACTION(S)

A rigorous justification for continued operation of Unit 1 was performed to allow for operation for one cycle following the clevis insert bolt failures.

AR# 2010-1804-18	Owner: ESY	Completion Date: 4/03/2010

A more rigorous justification for continued operation of Unit 1 was performed to allow for operation up to two cycles following the clevis insert bolt failures to allow development and implementation of a repair.

AR# 2010-1804-23	Owner: ESY	Completion Date: 8/19/2011

D.C. Cook Unit 2 Engineering Evaluation of the Radial Support System Clevis Insert Bolts and Operation through Spring of 2012.

AR# 2010-1804-22	Owner: DEM	Completion Date: 9/16/2010

Unit 2 LRSS Clevis Insert Inspection

WO# 55371767-39	Owner: EISI	Completion Date: 11/1/2010

Developed and implemented a minimum bolting pattern under EC-51640, "RX Vessel Lower Radial Support System (LRSS) Clevis Replacement Bolting for Unit 1," which incorporated an improved heat treatment and an under head radius that reduced peak stresses.

WO# 55399712	Owner: TPG	Completion Date: 9/28/2013

1.6.2 ADDITIONAL CORRECTIVE ACTION(S)

Provide root cause findings to the EPRI MRP and PWROG MSC to allow for evaluation and disposition of results for use in industry guidance.

CA# 2010-1804-34	Owner: DEM	Due Date: 4/2/2014

Update OE30993 based on results of Root Cause of failed bolts.

CA# 2010-1804-38	Owner: DEM	Due Date: 4/2/2014

1.6.3 ADDITIONAL ACTION(S)

WO package CREM comments for the U1C23 ISI inspection made no mention to the failure of the clevis insert bolts. Specifically CREM comments in WO 55343766-14 for the Lower Internal Visual Examination perform in U1C23 indicate inspection SAT despite failures observed on 7 LRSS clevis insert bolts and one dowel pin.

AR# 2013-19412	Owner: ENU	Due date: 2/20/2014

Consider the cost benefit of the enhancement of zinc addition to the RCS with respect to the potential to inhibit PWSCC.

GT# 2013-19422	Owner: CHM	Due Date: 1/29/2014

2.0 DETAILED REPORT

2.1 BACKGROUND

The design and function of the LRSS is described in Exhibit 8b and Exhibit 9 as follows:

In Westinghouse-style pressurized water reactors (PWRs), the LRSS represents the interface between the lower reactor internals and the reactor vessel (RV). The LRSS consists of six support locations equally spaced around the bottom circumference of the RV (60-degree spacing), see Figure 1 and Figure 2. Each support location consists of the following:

- A Radial Support Key (RSK) attached to the core barrel
- A clevis insert
- A vessel clevis lug, which is welded to the wall of the RV

See Figure 3 for the interface between the RSK and the clevis insert. For the D.C. Cook design, each core barrel clevis insert contains eight bolts (cap screws), for a total of 48 bolts. These bolts are restrained from rotation by a welded lock bar at each bolt location. In the event that a bolt fails, the lock bar design captures the bolt and keeps it in place to prevent a loose parts condition. Each insert location also contains 2 dowel pins, for a total of 12 pins. The clevis insert dowel pins provide added, possibly redundant, retention of the insert from long term vibratory motion. The interference fit of the pins keeps the insert from having small displacement slippage in the upward direction over time. They also help to prevent the clevis from sliding upward during core barrel removal. This prevents production of bending stresses in the bolt shanks if the clevis should shift. The clevis insert design and bolts prevent motion in other directions. Table 1 describes the bolt, lock bar, and dowel pin materials and sizes. Figure 4 illustrates the as-designed installation of the clevis inserts at D.C. Cook.

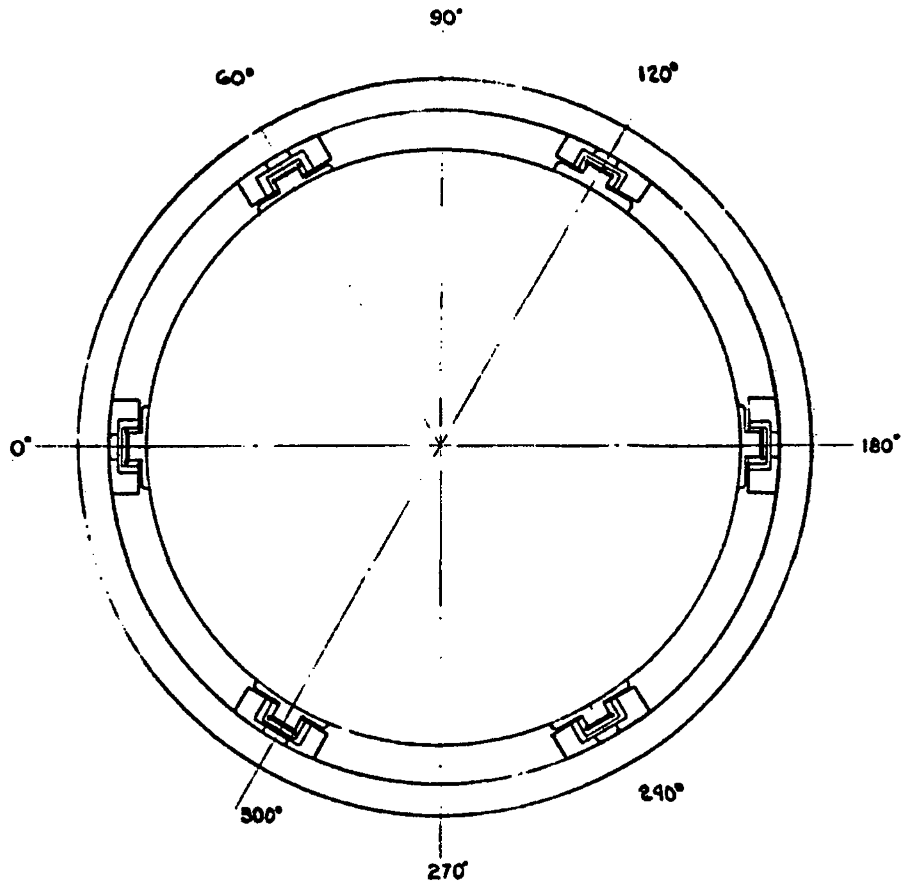


Figure 1: Plan view of LRSS

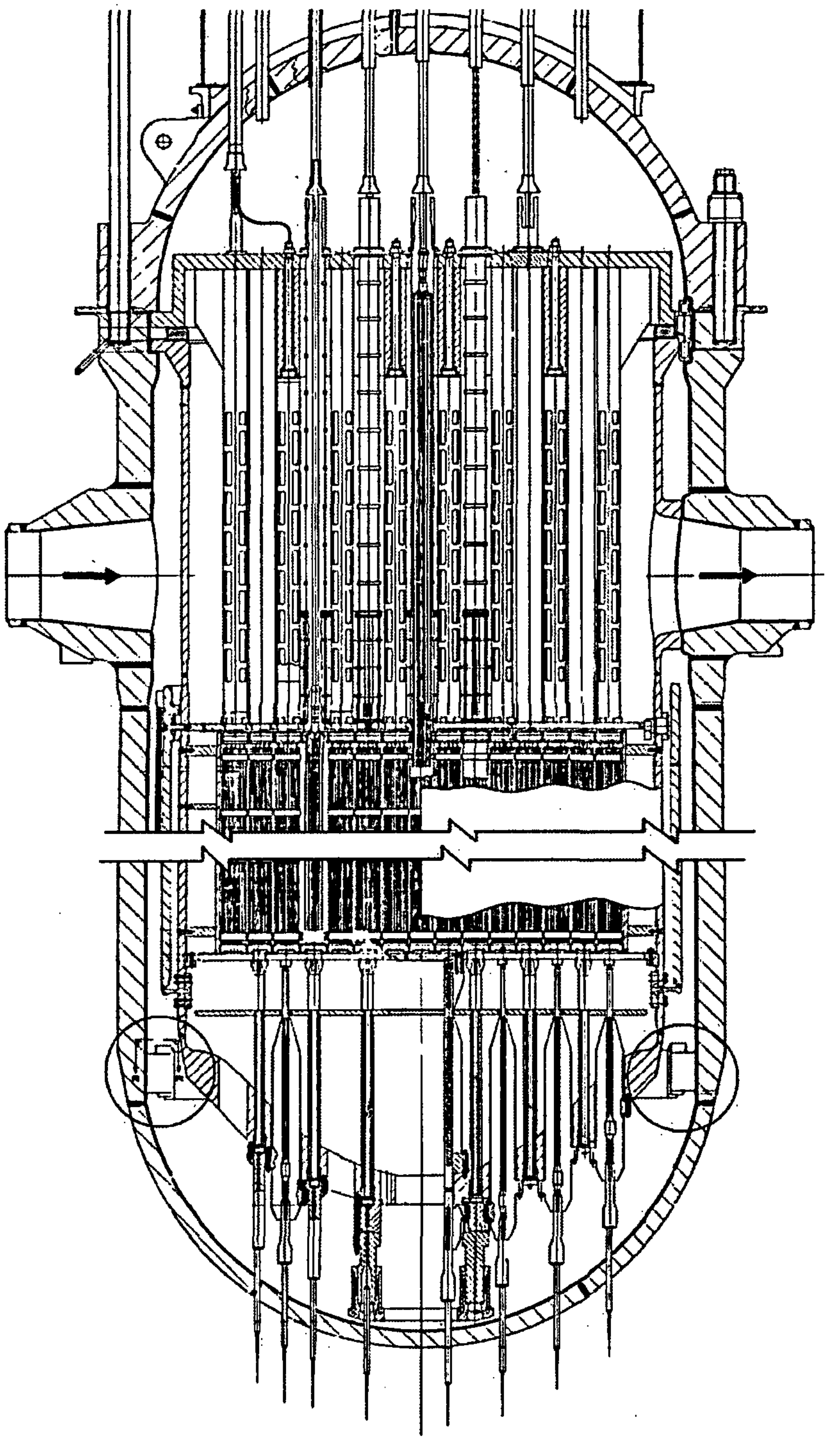


Figure 2: Elevation View of LRSS

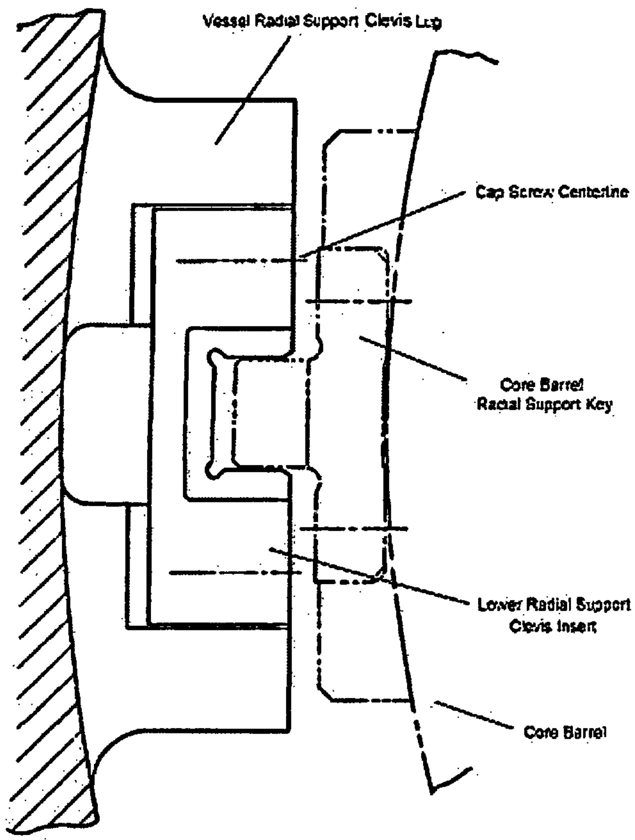


Figure 3: RSK and Clevis Insert Interference

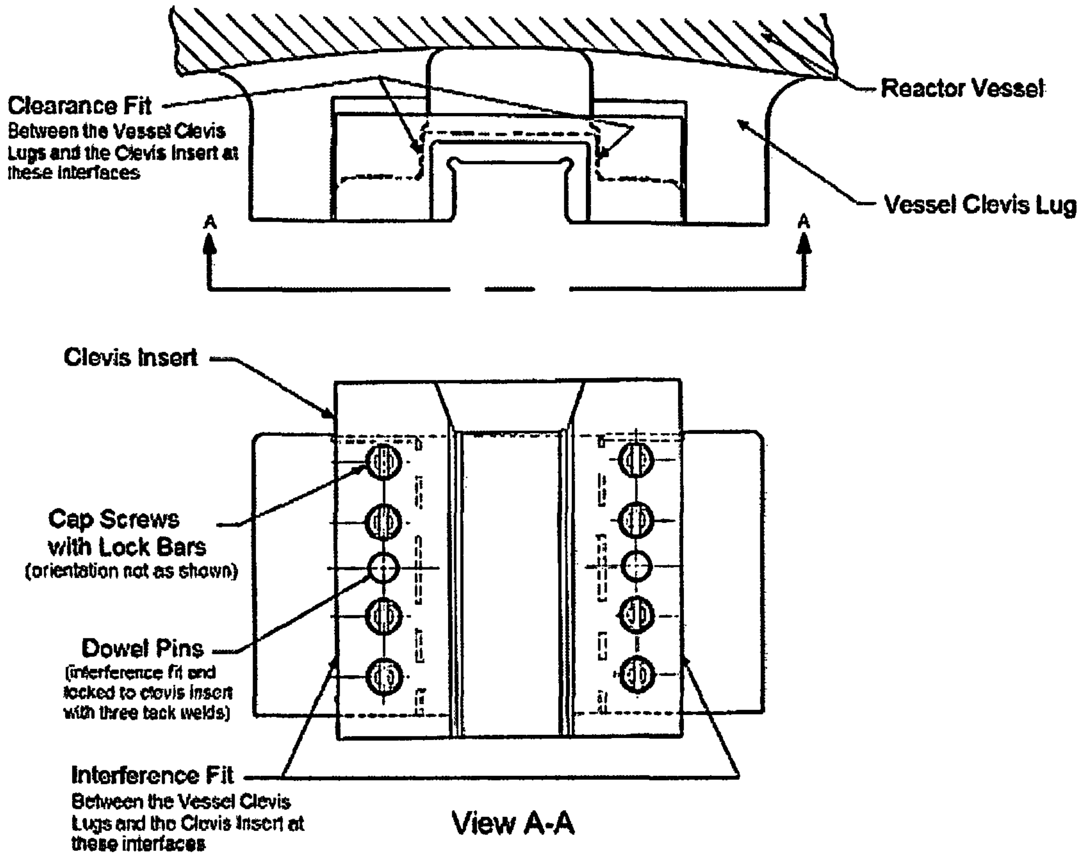


Figure 4: As-designed Clevis Insert Installation

Table 1: Materials and Sizes of Bolts, Lock Bar and Dowel Pin

Component	Material	Dimension
Clevis Bolts	Alloy X-750	Shank 2.50" 1-8 Thread Head 1.0" x 1.50" dia.
Lock Bar	ASTM B-166 (Ni-Cr-Fe) Annealed Bar	1.56 x 0.25 x 0.35
Dowel Pin	ASTM B-166 (Ni-Cr-Fe) Annealed Rod	3.75 x 1.375 dia.

2.2 DETAILED EVENT DESCRIPTION

On March 20, 2010, during the 10-year ISI of the D. C. Cook Unit 1 reactor vessel, an anomaly was noted concerning the LRSS clevis inserts. The 10-year ISI program includes a remotely operated visual inspection of the vessel after the core barrel is removed. This visual inspection examines the condition of the six clevis inserts, including general integrity of bolted and welded connections.

Damage was observed in the LRSS clevis insert bolts. Wear was observed on the lock bars and cap screw heads at a total of seven bolt locations; at five of these locations, the bolt heads were dislodged from their expected locations. Additionally, at one location, the three tack welds holding the dowel pin in place were noted to be broken. Figure 5 shows an image of a typical broken bolt (left) and the broken dowel pin tack welds (right). Table 2 shows the as-found condition of the cap screw and dowel pin with respect to each clevis location [Exhibit 9].

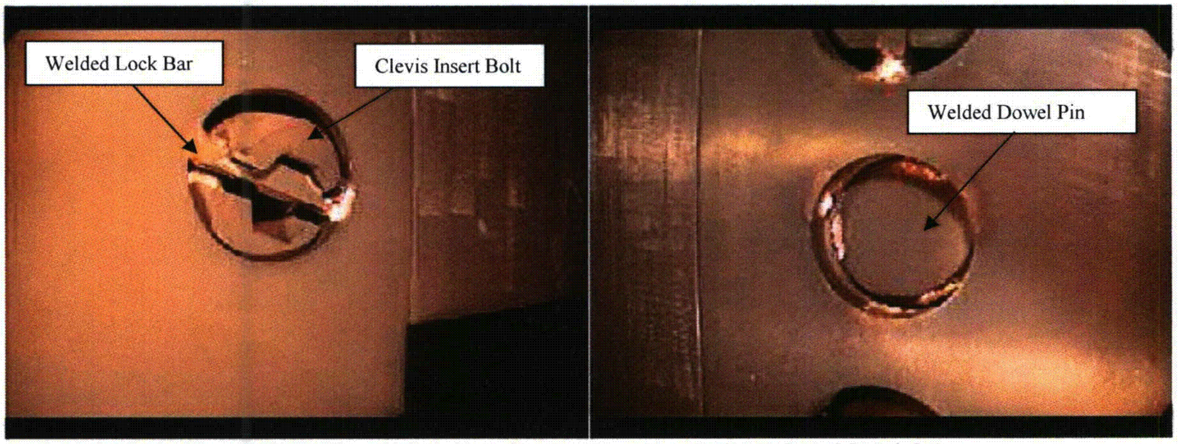


Figure 5: Typical condition of broken bolt (left). Broken dowel pin tack welds (right)

Following the discovery of the degraded clevis bolting, Westinghouse performed operability assessments allowing D.C. Cook Unit 1 to run until spring 2013 prior to replacement of the damaged clevis insert bolting. During Unit 1 Cycle 25 refueling outage in March 2013, a total of 29 clevis insert bolts were removed from the reactor vessel. Westinghouse performed a detailed evaluation defining a minimum acceptable bolting pattern on the clevis inserts for continued operation of Unit 1. The minimum acceptable bolting pattern, totaling 28 new clevis insert bolts, was installed during Unit 1 Cycle 25 refueling outage.

In September of 2010, Westinghouse performed an engineering evaluation of the Unit 2 LRSS clevis insert bolts, justifying continued operation of Unit 2 until inspections could be performed in spring 2012. During the Unit 2 Cycle 19 refueling outage October 2010, the core barrel was removed from the reactor vessel due to the discovery of baffle-former bolt degradation. With the core barrel removed, the decision was made to perform a VT-3 examination on the Unit 2 clevis insert bolts. No indication of degradation was observed on Unit 2 clevis insert bolting at the time of inspection.

Table 2: As-Found Unit 1 Cap Screw and Dowel Pin Conditions from March 2010

As-Found Cap Screw and Dowel Pin Conditions					
0° Location	**Left Side**	**Right Side**	**180° Location**	**Left Side**	**Right Side**
Top Bolt	(1)	(1)	Top Bolt	(1)	(1)
Second Bolt	(1)	(1)	Second Bolt	(1)	(1)
Dowel Pin	(1)	(1)	Dowel Pin	(1)	(1)
Third Bolt	(1)	(1)	Third Bolt	(1)	Wear/Dislodged
Fourth Bolt	(1)	(1)	Fourth Bolt	(1)	Wear/Dislodged
60° Location	**Left Side**	**Right Side**	**240° Location**	**Left Side**	**Right Side**
Top Bolt	(1)	(1)	Top Bolt	(1)	(1)
Second Bolt	(1)	(1)	Second Bolt	(1)	(1)
Dowel Pin	(1)	(1)	Dowel Pin	(1)	(1)
Third Bolt	(1)	(1)	Third Bolt	(1)	(1)
Fourth Bolt	(1)	(1)	Fourth Bolt	(1)	(1)
120° Location	**Left Side**	**Right Side**	**300° Location**	**Left Side**	**Right Side**
Top Bolt	Wear/Dislodged	(1)	Top Bolt	(1)	Wear
Second Bolt	(1)	(1)	Second Bolt	(1)	Wear
Dowel Pin	Cracked Welds and rotated 20-30° CCW(2)	(1)	Dowel Pin	(1)	(1)
Third Bolt	Wear/Dislodged	(1)	Third Bolt	(1)	(1)
Fourth Bolt	Wear/Dislodged	(1)	Fourth Bolt	(1)	(1)

Note:

1. No visible indication.
2. Counter clockwise.

2.3 EXTENT OF CONDITION

The extent of condition review was performed to identify where else the station was vulnerable to the same or similar condition. The condition was that LRSS clevis insert bolts and dowel pin failed unexpectedly. The immediate extent of condition includes the 19 LRSS clevis insert bolts in Unit 1 which were not replaced during the minimum bolt replacement in the Unit 1 Cycle 25 refueling outage and the 48 LRSS clevis insert bolts in Unit 2. This bounds the extent of condition.

The observed condition of bolt failures in Unit 1 was identified by the existing actions in place to perform visual inspections during each 10 year in-service inspection. The indications were documented and corrected through the Corrective Action Program. The actions taken were adequate to identify and correct the condition in Unit 1. There are no safety or operability concerns resulting from failure of LRSS clevis insert bolts or dowel pins. A rigorous loose parts generation and transport evaluation was performed to justify continued operation for two cycles following discovery. Therefore, no additional corrective actions are being suggested for Unit 1 or Unit 2 resulting from failure of the LRSS clevis insert bolts.

2.4 EVENT ANALYSIS

2.4.1 FAILURE ANALYSIS

Metallurgical Analysis: "Babcock & Wilcox Technical Services Group (B&W) Report" S-1473-002

During the Unit 1 Cycle 25 refueling outage in March/April 2013, a total of 29 clevis insert bolts were removed from the reactor vessel. Of the 29 bolts that were removed, 13 were considered intact (later found to be cracked) and 16 were fractured (head separated from shank). These bolts were sent to B&W for failure analysis. The failure analysis concluded the following [Exhibit 5e]:

- All of the 29 submitted bolts contained cracking in the head-to-shank transition; no cracking was identified in the threaded region of any bolts.
- There was a generally uniform fracture pattern observed in the bolts, which consisted of crack initiation at two diametrically opposing sides of the bolt in the head-to-shank transition region and crack growth that extended upward into the bolt head at a ~35° angle relative to horizontal.
- Fractographic SEM analysis and cross section metallographic examinations confirmed the fracture mode was essentially 100% intergranular on all of the bolts. Very minor mixed mode cracking consisting of transgranular cleavage and ductile fracture was noted near the center of one bolt which would have been the final failure region.
- There was no evidence that the bolts failed due to fatigue cracking or mechanical overload.
- The chemical analysis results for all four bolts were consistent with Alloy X-750 material.
- The mechanical properties and microstructure of the bolts were consistent with those published for Alloy X-750 material.
- No unexpected characteristics in the material properties, microstructures, or form of the bolts were identified.
- **The laboratory data indicated the bolts failed by intergranular stress corrosion cracking (IGSCC).**

2.4.2 EVENT TIMELINE

A Timeline was created to depict the chronology of relevant events since initial construction of DC Cook Units 1 and 2 to support the detailed event description provided in Section 2.2. The Timeline focused on the following:

- Inspections and results
- Dates of published industry guidance for reactor vessel internals
- The event itself
- The interim actions that were put into place following the event
- Time periods with conditions that could promote IGSCC

See Attachment 5.3.

2.4.3 FAILURE MODES AND EFFECTS ANALYSIS

A Failure Modes and Effects Analysis was performed to identify all possible failure modes of the LRSS clevis insert bolts and dowel pin failures through a collegial review of the identified problem and third party evaluation, as seen in Attachment 5.4. The effects from the identified failures were then used to build conclusions in the Support Refute Matrix discussed below and shown in Attachment 5.5.

2.4.4 SUPPORT REFUTE MATRIX

A Support Refute Matrix was created to support or refute the failure modes identified in the failure modes and effects analysis and determine which of them most likely caused or contributed to the LRSS clevis bolts and dowel pin failure (See Attachment 5.5). The following were supported in the support/refute matrix for the clevis bolt failures:

- Intergranular Stress Corrosion Cracking (IGSCC)/Primary Water SCC (PWSCC)
  - Improper material heat treatment specification
  - Inadequate bolt design dimensions at the bolt head to shank radius
- Low Temperature Crack Propagation (LTCP) – Potential contributor

IGSCC/PWSCC requires a combination of tensile stresses (both applied and/or residual), a corrosive environment, and a susceptible material to be present. IGSCC/PWSCC will not occur if any of these three factors is eliminated. The support refute found that the stresses produced at the head to shank radius were not sufficient to cause overload failure of the LRSS clevis insert bolts. However, due to the tight dimensions at this location the stresses produced were found to be a contributing cause to PWSCC. The support refute also found that the heat treatment used on the LRSS clevis insert bolts is now well known to be susceptible to SCC in a PWR environment. Without the use of this susceptible heat treatment SCC would not have occurred.

These potential failure modes are discussed in more detail in subsequent sections of this evaluation.

2.4.5 WHY STAIRCASE

A Why Staircase was constructed to validate the causal factors leading to failure of Unit 1 Clevis Insert Bolts. (See Attachment 5.6) Completion of the Why Staircase validated the following root cause:

Root Cause: Primary water stress corrosion cracking (PWSCC) of the clevis insert bolts due to the use of Alloy X-750 with a susceptible heat treatment.

Stress corrosion cracking is a known phenomenon that occurs in Alloy X-750 nickel alloy components under specific conditions. Figure 6 below shows the conditions that need to be present for SCC to occur. SCC will not occur if any of these three factors is out of range of susceptibility. In the case of the D.C. Cook Unit 1 clevis insert bolts, the bolts failed intergranularly (at the grain boundaries of the metal); thus the failure mechanism being IGSCC. Alloy X-750 bolts have historically failed by IGSCC in PWR primary water environments; a phenomenon also referred to as Primary Water Stress Corrosion Cracking (PWSCC). [Exhibit 42]

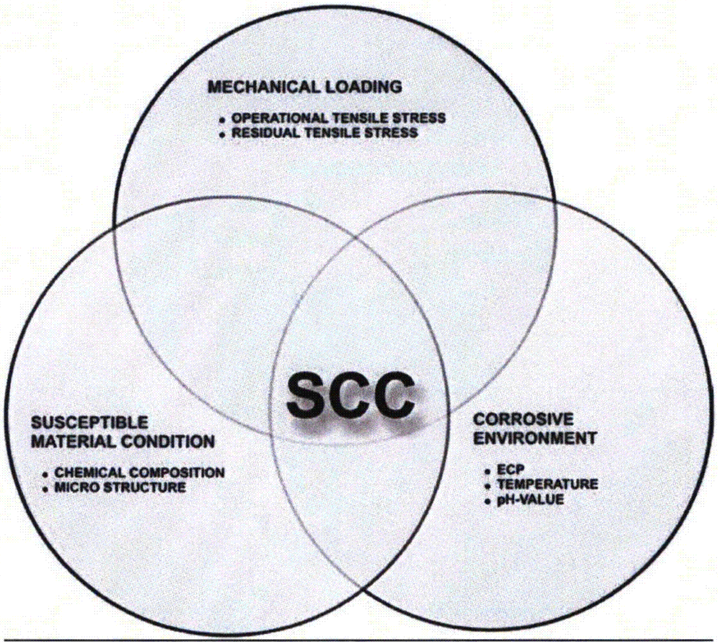


Figure 6: Synergistic Effects Required for Stress Corrosion in Metals

Material and Heat Treatment Selection of Alloy X-750 [Ref: Exhibit 12]

Nickel based Alloy X-750 was originally developed for use in high temperature applications where good corrosion resistance and high strength are required, such as in gas turbines. Alloy X-750 is typically used in nuclear power plants where high corrosion resistance similar to that of Alloy 600 is required and higher strength and fatigue resistance is needed, such as in control rod guide tube support pins. Alloy X-750 is now considered a 'mature' alloy in that there has been extensive research, testing, and improvements of the alloy. As discussed in the EPRI Materials Handbook for Pressure Boundary Applications, perhaps the most important result of industry research on Alloy X-750 was the determination that the heat treatment condition strongly affects its susceptibility to SCC in PWR and BWR environments. It was found that heat treatments used in early industry specifications, while suitable for non-Light Water Reactor environments such as gas turbines, resulted in high susceptibility to SCC in high temperature water environments.

Heat treatment of Alloy X-750 typically starts with a hot worked material that is solution heat treated at a high temperature to ensure a single uniform microstructure is present prior to aging. Typical heat treatments for Alloy X-750 components used in early nuclear plant construction included the AH and BH heat treatments which call for solution treating at 1625 °F and 1800 °F respectively. There is a significant amount of operating experience showing failures of Alloy X-750 bolts in this heat treatment when exposed to primary water at high temperatures [Exhibit 12]. Current industry guidance shows a significant improvement in resistance to stress corrosion cracking when the solution treating temperature is raised to 2000 °F such as with the HTH heat treatment. The LRSS clevis insert bolts that failed at CNP Unit 1 were manufactured in accordance with Westinghouse Materials Specification 70041 EJ, which calls for an equalization heat treatment of 1625 °F for 24 hours followed by a solution treatment of 1775 °F for one hour. Therefore, the D.C. Cook Unit 1 clevis insert bolts were heat treated using a process that is known to make Alloy X-750 susceptible PWSCC, which was identified as the root cause to the observed PWSCC.

Root Cause: Primary water stress corrosion cracking (PWSCC) of the clevis insert bolts due to the use of Alloy X-750 with a susceptible heat treatment.

<u>High Stress at Head to Shank Radius</u>

The B&W metallurgical analysis determined that all removed LRSS clevis insert bolts failed or exhibited cracking in the head to shank transition region [Exhibit 5e]. The bolt head to shank transition or radius acts as a stress riser and is one of the areas of highest localized stresses on the bolts. A high stress concentration can result in high peak stresses depending on the preload and any thermal or operating stresses.

MRP-175, "PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values," gives screening criteria for aging degradation mechanisms. In the case of Alloy X-750, the document sets a stress corrosion cracking screening criterion of 100 ksi or greater for Alloy X-750 in the HTH condition. An Alloy X-750 component loaded beyond this level has some finite probability of experiencing SCC. MRP-175 also notes that Alloy X-750 in the AH or BH heat treated condition (similar heat treatments to that of the CNP LRSS clevis insert bolts) would be even more susceptible than material in the HTH condition and would thus have an even lower screening criterion. According to a Westinghouse stress evaluation, the bolts were subject to as much as 95 ksi in the head to shank transition due to preload and thermal stresses while all of the bolts were intact [Exhibit 58 and Exhibit 8b]. This stress is high enough for stress corrosion cracking initiation of the more susceptible Alloy X-750 material at D.C. Cook. Hence the following was identified as a contributing cause to the observed PWSCC.

Contributing Cause: Localized peak stresses at the LRSS clevis insert bolt head to shank radius due to inadequate bolt dimensions.

PWR Primary Water Environment

There is extensive operating experience for Alloy X-750 bolts with a susceptible heat treatment failing in primary water by PWSCC [Exhibit 12]. EPRI guidelines for stress criteria described above are based on maintaining a water chemistry condition in accordance with EPRI PWR Primary Water Chemistry Guideline. Any anomalies in the Primary Water Chemistry at D.C. Cook Unit 1 could explain why the bolts in Unit 1 failed when no other failures have been reported throughout the industry [Exhibit 42].

D.C. Cook Chemistry Department personnel indicate that the CNP primary chemistry program complies with EPRI's Pressurized Water Reactor Primary Water Chemistry Guidelines.

Technical Requirements Manual limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion cracking. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. [TRM 8.4] Previous Technical Specification primary chemistry limits and bases were similar.

Historical Licensee Event Reports for out-of-specification primary chemistry were reviewed with chemistry personnel and determined to be not applicable to PWSCC [Exhibits 118 and Exhibit 119].

Review of PWR chemistry recommendations found that RCS zinc addition could be a possible inhibitor to PWSCC initiation and propagation in nickel-based alloys. Approximately 30 percent of PWRs currently utilize zinc addition [Exhibit 132]. RCS zinc addition was most recently evaluated at CNP for dose reduction (GT 00844243-35) and rejected. As an enhancement, a general tracker (GT 2013-19422) has been created to consider the cost benefit of zinc addition to the RCS with respect to the potential to inhibit PWSCC.

2.4.6 LOW TEMPERATURE CRACK PROPAGATION (LTCP)

LTCP is a driving mechanism for crack growth that has been seen in Alloy X-750 materials. Industry literature has shown LTCP occurs under the following conditions [Exhibit 43]:

- Temperatures between 122 and 302 degrees F
- Hydrogen concentrations > 20cc H2/kg H2O
- Existing crack like defect to act as an initiation site

When these conditions are present, Alloy X-750 can experience rapid cracking (rates of crack propagation on the order of millimeters per minute) due to a hydrogen embrittlement effect. Hydrogen can come externally from the coolant in contact with the metal, from corrosion of the metal surface or it can be present internally due to previous exposure to hydrogen. Chemistry records from the Unit 1 extended shutdowns in 1998-2000 and 2008-2009 were reviewed and it was determined that periods existed in which temperature and hydrogen conditions would promote LTCP during the first approximate six days of the 1998-2000 extended shutdown. For the 2008-2009 extended shutdown, hydrogen levels were reduced to below 20cc H_2/kg H_2O on the first day of the turbine event and coming out of this shutdown period temperatures were raised above 302 degrees F prior to hydrogen levels elevating back above 20cc H_2/kg H_2O. [Exhibit 8b]

The failure analysis performed by B&W found no indications of existing crack-like defects or evidence of machining or fabrication issues that could have led to LTCP of the clevis insert bolts. Therefore, if LTCP played a role in the failure of the LRSS clevis insert bolts, it would have only been a driver once cracking initiated from PWSCC. Preventing LTCP would not have prevented crack initiation or the eventual fracture of the bolts. Due to the rapid crack growth rates associated with LTCP, and without knowledge of when the bolt heads failed, LTCP cannot be validated as a contributing cause.

Further review of the CNP chemistry controls was conducted by the team to determine if additional corrective actions would be warranted to reduce the time spent in LTCP conditions. During normal operation LTCP is not of concern due to the RCS temperature being above the temperatures at which LTCP would occur. 12-THP-6020-CHM-110, "RCS Chemistry - Shutdown and Refueling" Figure 1 provides the RCS dissolved hydrogen target bands. Review of this figure shows that actions are already in place to ensure that hydrogen levels stay below 20cc H_2/kg in the temperature range of concern for LTCP. It appears that these controls were put in place during revision 8 of 12-THP-6020-CHM-110 in 2002 [Exhibit 125]. No additional actions are needed. See Exhibit 8b & Exhibit 43 for further information on LTCP.

2.4.7 DOWEL PIN

The following description of the dowel pin is modified from Exhibit 8b:

The Alloy 600 dowel pins were interference fit into the clevis insert and lug during construction. The tack welds were intended to provide capture of the pin in case of failure and provide no structural purpose. The welds are small and offer relatively little resistance to pin motion as compared to the resistance resulting from the interference fit. There are no obvious indications of deformation in the welds as observed from inspection images. Therefore, the most likely cause of failure is either fatigue or SCC. The forces required to rotate and translate the pin, as observed in the ISI inspection, would be more than sufficient to drive either failure mechanism.

Normal loading of the interference fit dowel pin would not be expected to break tack welds and cause rotation or translation of the pin. It is significant that the dowel pin with broken tack welds and movement was located on the clevis insert where the most bolt damage was observed. All of the bolts on this clevis had completely fractured. The loss of clamping load on this side of the clevis with the displaced dowel pin may have resulted in vibration associated with the cyclic loading of the insert during operation. The dowel pin would then be subject to vibration due to loss of clamping load.

Normal loading of the clevis insert following bolt failure was likely sufficient to drive rotation and axial translation of the dowel pin. The dowel pin must not have been fully seated in the reamed hole at installation which allowed the pin to be driven deeper into the lug. The dowel pin holes were reamed after installation of the clevis insert, so any Poisson burr [1] remnant from machining would tend to ratchet movement of the dowel pin deeper into the hole. Therefore, the failure of the dowel pin tack welds and dowel pin motion is a consequence of the bolt failure at that location.

Figure 7 shows an overview of the clevis insert and the location of cross section for the following two figures. Figure 8 shows an exaggeration of how the clevis insert would deflect under radial loading with intact bolts. Figure 9 shows an exaggeration of how the clevis insert would deflect under radial loading with failed bolts [Exhibit 94].

[1] Plastic deformation of material which may be caused by machining operations such as edge features at the exit face of material after drilling or reaming.

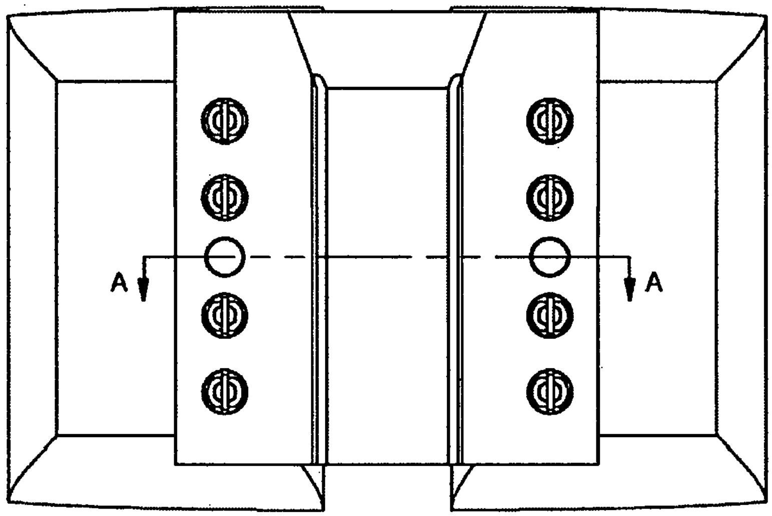


Figure 7: LRSS Clevis Insert and Lugs

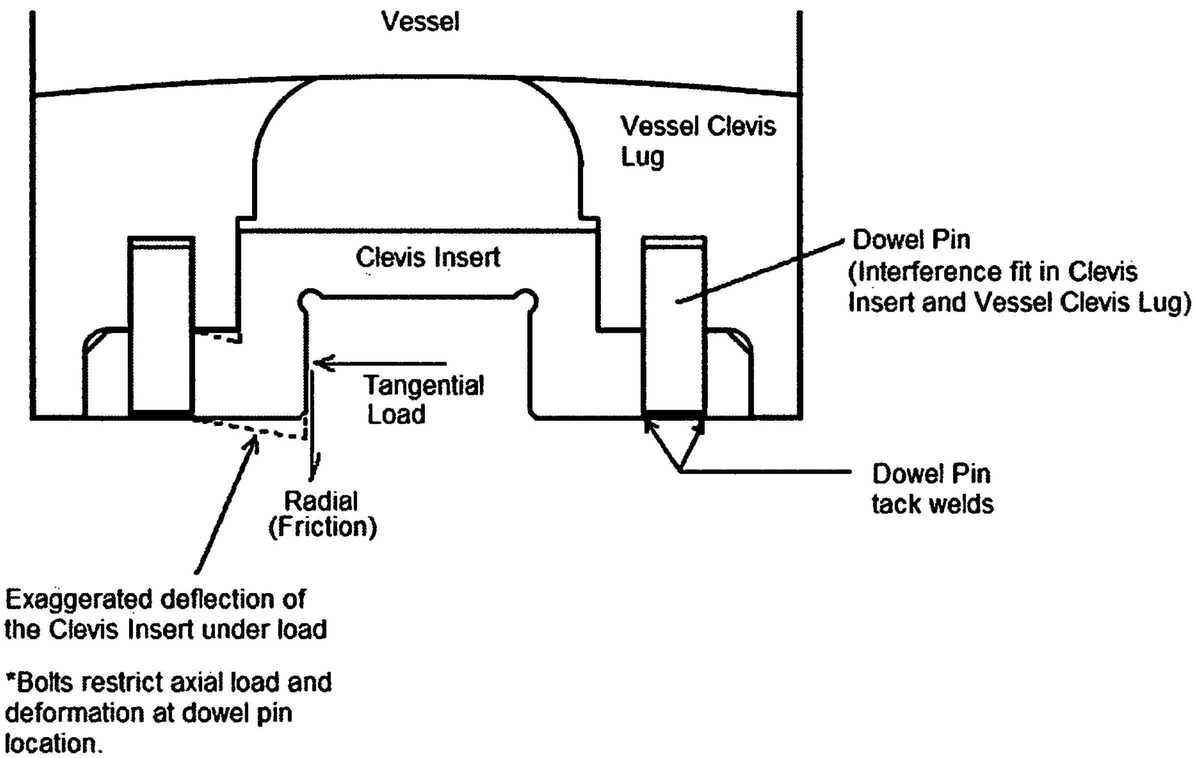


Figure 8: Bolts Intact on Load Side Section A-A

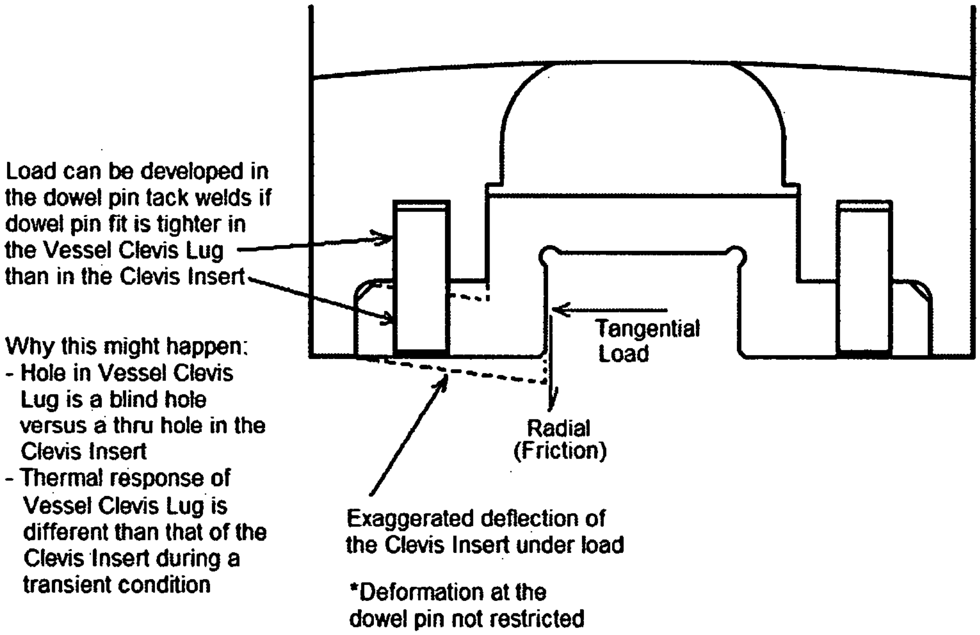


Figure 9: Bolts Broken on Load Side Section A-A

2.4.8 INDUSTRY GUIDANCE

Because CNP Units 1 and 2 have Alloy X-750 LRSS clevis insert bolts with a heat treatment that has extensive operating experience showing it produces components susceptible to PWSCC, the root cause team reviewed the aging management plan and industry guidance for managing the aging of LRSS clevis insert bolts. The conclusions of this review found that industry guidance indicated a medium likelihood of stress corrosion cracking of clevis insert bolts. However, no actions beyond inspecting per the 10-year ISI were recommended due to overall effect of PWSCC of the clevis insert bolts on the clevis insert not being significant. The industry guidance documents reviewed found that the existing 10 year in-service inspection or other in-place aging management plans are sufficient to preclude a safety, reliability, or financial concern. A closer review of the guidance document attributes and their relation to the Reactor Vessel Internals Aging Management Program is provided in Attachment 5.7. Furthermore, the event and causal factor chart (Timeline) shown in Attachment 5.3 lists the industry guidance documents that the CNP reactor vessel internals aging management program is based on and when they were published.

2.4.9 EVALUATION OF INSPECTION METHOD AND FREQUENCY

The issues reviewed in this evaluation supported the 10 year ISI VT-3 examination as the appropriate periodicity and inspection based on the low safety and operability significance of clevis bolt failure. The reactor vessel internals aging management guidance document MRP-227-A, shows the clevis insert bolts are considered to have a low likelihood of damage, medium likelihood of failure, and a consequence of failure being only a significant economic impact. This guidance document has been updated since the observed LRSS clevis bolt failures and the event is included in the OE.

The condition reviewed in this evaluation did not result in the loss of the ability of the LRSS to perform its intended design function. Even with the postulated failure of all bolts, all dowel pins, and clevis-to-lug interference fit, the clevis insert will remain in place due to the capturing geometry of the vessel lugs, core barrel radial keys, and the bolt and dowel pin remnants. The LRSS will maintain its design function in all operational and faulted conditions. [Exhibits 6, 8b, 20, 55, 74, 85a, 87b, 88, 113, and 121]

2.5 ORGANIZATIONAL AND PROGRAMMATIC FAILURE MODE REVIEW

This evaluation found no organizational or programmatic issues. All evidence shows that the clevis insert bolts were known to have a potential for failure, but because of the low significance of failure from a safety and operational standpoint, and no history of failure in the industry, no contingencies were in place in the event of a failure. Existing industry guidance found in MRP-227 relied on the ASME Section XI 10-year ISI inspection for ensuring the integrity of the clevis insert bolts, which properly identified the failure. See Attachment 5.8 for Organizational And Programmatic Failure Mode Review.

2.6 SAFETY CULTURE IMPACT REVIEW

A Safety Culture Impact Review was performed looking at Nuclear Regulatory Commission (NRC) Components of Safety Culture and Institute of Nuclear Power Operations (INPO) 8 Principles of a Strong Nuclear Safety Culture. This evaluation found no applicable cross-cutting area components or nuclear safety culture issues related to the failure of the LRSS clevis bolts. See Attachment 5.9 for Safety Culture Impact Review.

2.7 EQUIPMENT RELIABILITY REVIEW

- Incorrect Classification (AP-913 INCORT) was applied due to the clevis bolts being incorrectly classified; the classification of the reactor vessel, 1-OME-1, which is a critical component, was applied. The LRSS clevis bolts are piece parts, not a piece of equipment. Therefore, they are not classified as critical, non-critical, or run-to-failure.

- Original Design Less than Adequate (AP-913 ORIG) was applied due to the bolts failing by PWSCC. The original design used an Alloy X-750 bolt in a heat treatment unknowingly susceptible to PWSCC. Additionally, all of the bolts failed in the head to shank transition due to the localized high stresses in this region of the bolt. The replacement bolts utilized an Alloy X-750 bolt with a less susceptible heat treatment and an improved head-to-shank design transition to reduce stress in this region.

See Attachment 5.10 for Equipment Reliability Review.

2.8 SAFETY SIGNIFICANCE

The condition described in this action was reviewed for its actual and potential impact on nuclear, radiological, personnel/industrial safety, and plant operation. Based on this review, the condition described is of low safety significance.

Nuclear Safety

The condition reviewed in this evaluation did not result in the loss of the ability of the LRSS to perform its intended design function. Even with the postulated failure of all bolts and dowel pins, clevis inserts are predicted to remain in place. [Exhibit 55 and Exhibit 8b]

Although bolt failures do not challenge the ability of the LRSS to perform its design function, it is important to the plant from a commercial perspective. The clevis inserts have several redundant means of attachment including bolts, pins, an interference fit into lugs, and the geometry of the assembled system. However, if the first three listed means of attachment are defeated, this creates a commercial concern. When the core barrel is removed for maintenance, a clevis insert may then be allowed to displace from the lugs and fall to the bottom of the reactor vessel. LRSS clevis replacement following displacement of a clevis insert from its lugs may challenge the economic viability of the plant. This significant clevis insert displacement cannot occur with the core barrel installed. The core barrel is not removed unless the reactor is completely defueled.

Radiological Safety

There was some impact on radiological safety. Dose for Unit 1 LRSS repairs was 4635 mR and there was one, level one personnel contamination.

Personnel/Industrial Safety

There was no impact on personnel or industrial safety from the condition reviewed in this evaluation.

Operational Impact

Plant operation with the condition described in this action had no adverse effect on the design functions of any UFSAR-described SSCs.

Probabilistic Risk Assessment (PRA)

The PRA group was contacted regarding a PRA model for the LRSS clevis insert bolts. The LRSS clevis insert bolts do not have a PRA model because PRA only considers SSCs where failures could cause a safety or operability concern. Therefore it is not appropriate to include them in a PRA model.

2.9 EVENT CONSEQUENCES

This event did not adversely impact operability or safety of plant operation. The event reviewed in this evaluation resulted in financial consequences due to operability evaluations, LRSS repair project, and impact on the U1C25 refueling outage schedule, duration, and dose. This event resulted in an ongoing high level of interest in the issue and our response to the issue from other utilities, industry groups, and the NRC.

3.0 OPERATING EXPERIENCE

The purpose of Operating Experience (OE) is to reduce the possibility of events occurring that have been previously identified by taking actions to improve the barriers to prevent occurrence, and benchmarking other utilities to identify actions taken that have been successful in preventing similar events.

A review of internal and external OE was conducted to identify events that had similar causes to the events that were investigated in this root cause evaluation. These OE were reviewed to identify potential corrective actions, contributing causes and lessons learned that could prevent future occurrences or assist in the analysis section of this report. The details of this review are documented in Attachment 5.11 of this report.

3.1 INTERNAL OPERATING EXPERIENCE

Searches of the ActionWay and the Cook OE database were conducted using the following key words:

- Split Pins
- Alloy X-750
- Clevis Bolts
- Lower Radial Support System (LRSS)
- Primary Water Stress Corrosion Cracking (PWSCC) & Bolt
- Low Temperature Crack Propagation (LTCP)

The majority of OE found in ActionWay was related to Alloy X-750 valve discs which were determined to be non-applicable to this evaluation. Additionally, there were several actions related to the control rod guide tube support pings (split pins) as discussed below. Searches also found actions related to PWSCC susceptibility of RCS piping which has previously been mitigated by the Cook Plant. The following actions were found to be most applicable to this evaluation.

GT 2012-1808/2012-1809

This GT LRP is to replace the Control Rod Guide Tube (CRGT) Support Pins (split pins) no later than U2C26 in Spring 2021 and U1C28 in Fall 2017.

The support pins align CRGTs to the upper core plate. The original material of the split pins was Alloy X-750 with a low temperature solution anneal heat treat. These pins were susceptible to PWSCC and failed. Split pins were replaced by Babcock & Wilcox Company in both D.C Cook Unit 1 and Unit 2 in the mid 1980's with new Alloy X-750 split pins with improved design to reduce stress risers and a high temperature solution anneal heat treat. These improved features reduced susceptibility to PWSCC, but did not eliminate the problem. The comparison of the split pin failure to the clevis bolts was discussed by Westinghouse in WCAP-14577 Rev 1-A, March 2001. While it was known that the Alloy X-750 clevis bolt material was susceptible to PWSCC, the failure was not considered likely due to differences in fluence, temperature, and stresses between the two bolts.

AR 2010-10940

This was the root cause evaluation for the core baffle bolt failures in the Unit 1 reactor vessel. The root cause of failed baffle-former bolts was Irradiation Assisted Stress Corrosion Cracking (IASCC) in conjunction with thermal and irradiation induced loss of preload in several baffle-former bolts. IASCC was considered in the Support/Refute analysis for this evaluation.

3.2 EXTERNAL OPERATING EXPERIENCE

Searches of the INPO OE database were conducted using the following key words:

- Split Pins
- Alloy X-750
- Clevis Bolts
- Lower Radial Support System (LRSS)
- Primary Water Stress Corrosion Cracking (PWSCC) & Bolt
- Low Temperature Crack Propagation (LTCP)

The results of the searches varied from no results to numerous results, depending on the key word search and format. A significant amount of OE was reviewed related to CRGT split pin failures of Alloy X-750 bolts by PWSCC. External OE related to split pin failures has not been included in this report considering the extent of internal OE at Cook. See internal operating experience in Section 3.1 of this report for D.C Cook related OE on split pin failures. The most applicable remaining results to this evaluation are detailed below.

NUREG 1801, Revision 2

The Generic Aging Lessons Learned (GALL) Report from the NRC identifies the clevis insert bolts as having a possible aging effect/mechanism as loss of material due to wear. This report supports the use of EPRI MRP-227 for managing the aging effects of clevis insert bolts.

OE99 – 009452 - Nine Mile Point Unit 1

During the 1999 refueling outage fifteen (RFO15), one of the Alloy X-750 3/8" cap screws was found to be broken during a visual inspection of a core shroud repair tie rod assembly. The root cause was determined to be IGSCC in conjunction with large, sustained differential thermal expansion stress due to fastening of dissimilar materials with the cap screw. The difference in thermal expansion between dissimilar materials being bolted together has been evaluated in the support/refute section of this report.

3.3 REPEAT EVENT OR FAILURE TO LEARN FROM PREVIOUS OPERATING EXPERIENCE

Review of the external and internal OE determined that the issues found in this root cause did not represent a repeat event and could not have been prevented by lessons learned in the previous events cited in this section.

Station personnel were aware of the X-750 LRSS clevis insert bolts, and their possible susceptibility to PWSCC, prior to observing bolt failures. The station followed inspection and aging management guidance provided in ASME Code, EPRI-MRP, and PWROG documents which indicated low risk and low consequence of failure.

3.4 LESSONS LEARNED

Due to the low significance of a failure and no history of failure in the industry, no contingencies were in place in the event of a failure of the clevis insert bolts. Existing industry guidance in MRP-227 relied on the ASME Section XI 10-year ISI inspection for ensuring the integrity of the clevis insert bolts. Failure mechanisms identified during the OE review have been added to the support/refute matrix for further evaluation.

4.0 EFFECTIVENESS REVIEW PLAN

No additional corrective actions to correct or preclude repetition of the observed condition were generated beyond those already performed. The ISI program continues to monitor the LRSS according to ASME Code. Therefore, no effectiveness review is required. This was discussed and resolved in the same meeting on January 20, 2014 that corrective actions were discussed. Attendees are listed in Section 1.3.1.

5.0 ATTACHMENTS

5.1 PRE-ANALYSIS

5.2 EXTENT OF CAUSE EVALUATION

5.3 EVENT TIMELINE

5.4 FAILURE MODES AND EFFECTS ANALYSIS

5.5 SUPPORT REFUTE MATRIX

5.6 WHY STAIRCSE

5.7 INDUSTRY GUIDANCE REVIEW FOR CLEVIS INSERT BOLTS

5.8 ORGANIZATIONAL AND PROGRAMMATIC FAILURE MODE REVIEW

5.9 SAFTETY CULTURE IMPACT REVIEW

5.10 EQUIPMENT RELIABLITY REVIEW

5.11 APPLICALBE OE

5.12 DOCUMENTS REVIEWED

5.13 CARB COMMENTS

5.1 PRE-ANALYSIS

Event Description

During the 10-year in-service inspection of the Donald C. Cook Nuclear Plant Unit 1 reactor vessel in March 2010, an anomaly was noted concerning the Lower Radial Support System (LRSS) clevis inserts. Seven (of 48) bolt locations had wear observed on the lock bar, indicating that the cap screw head had detached from the shank and that flow was causing the head to vibrate against the lock bar. One (of 12) dowel pin had broken tack welds and it was rotated and displaced into the clevis insert.

Problem Statement

Lower Radial Support System (LRSS) clevis insert bolts and dowel pin failed unexpectedly resulting in the organization addressing emergent concerns of structural margin and loose parts in the reactor coolant system.

Prompt and Interim Actions

A rigorous justification for continued operation was generated for two cycles to allow for repair development. A minimum bolt pattern was installed per EC-51640, "RX Vessel Lower Radial Support System (LRSS) Clevis Replacement Bolting for Unit 1."

Scope of Evaluation

The evaluation will investigate the causes and potential organizational and programmatic issues regarding the LRSS bolt and dowel pin failure through the use of metallurgical testing, support/refute analysis, failure modes and effects analysis, and why staircase.

Management Sponsor, Team Lead, and Team Members

Management Sponsor:	April Lloyd (AEP/DE)
Team Lead:	Alex Olp (AEP/DEM)
Cause Evaluator:	Lloyd Burton (AEP/WCA)
PID Mentor:	Jessica Huycke (AEP/PID)
Team Members:	Kevin Kalchik (AEP/DEM)
	Matthew White (AEP/ENU)
	James Greendonner (AEP Operations)
	Ben Blumka (AEP Maintenance)
	Kevin Neubert (Westinghouse)
	Gary Thompson (MPR)

Schedule

Present Pre-analysis to CARB:	10/25/13
Team Pre-job Brief:	11/13/13
CARB Update:	12/6/13
Evaluation Approval:	12/20/13

5.2 EXTENT OF CAUSE EVALUATION

The bolts failed because of PWSCC of the clevis inset bolts due to the use of Alloy X-750 with a susceptible heat treatment. Other nickel based alloys could be susceptible to stress corrosion cracking in primary water under tensile stress.

This evaluation is bounded based on several factors. The initial volume of interest was unisolable portions of the RCS. There are many programs which cover inspection and aging management of the system and specific components within the system. The clevis bolt failures occurred in the reactor vessel, so the review is limited to this volume because this presents a higher consequence for non-pressure boundary component failures for safety, operability, and economic impact due to difficulties associated with repair/replacement activities.

The scope of the evaluation does not include consumable items, such as fuel assemblies, reactivity control assemblies, or nuclear instrumentation, because these components are not typically within the scope of the components that are required to be subject to an aging management review, as defined by the criteria set in 10 CFR 54.21(a)(1). Therefore, the investigation is bounded to non-consumable components manufactured with nickel based alloys in the volume of the reactor vessel and internals exposed to primary water that do not have a pressure boundary function.

The reactor vessel is governed by multiple programs which include, but are not limited to, the RVI AMP and the Reactor Vessel Integrity Program. The RVI AMP typically governs reactor vessel internal removable structures, not all non-pressure boundary features and components inside the reactor vessel. The Reactor Vessel Integrity Program manages the reduction of fracture toughness due to irradiation embrittlement of reactor vessel beltline materials to assure that the pressure boundary function of the reactor vessel beltline is maintained. The LRSS clevis inserts, bolts, dowel pins, and lugs seem to fall at the interface between these two particular major component groups and programs.

Table 3 of Attachment 5.2 is a summary of nickel based alloy components in the reactor vessels contained in MRP-191, a document created during the development of MRP-227-A to screen, categorize, and rank RVI components. The last two columns discuss any additional relevant details or management strategies specific to DC Cook for these components.

The ASME Section XI ISI Program includes, but is not limited to, inspections of the reactor vessel internals, and the reactor vessels including its internal components and attachments. This program is credited in other programs, including the RVI AMP, for VT-3 visual inspections of LRSS components, including clevis insert bolts.

The Alloy 600 Material Management Program manages aging effects of both pressure and non-pressure boundary components made of Alloy 600/690, 82/182, and 52/152 materials in the RCS. The clevis inserts, lugs, and lug welds are considered and evaluated in this program. However, the clevis bolts are not included in the Alloy 600 Program because they are fabricated from Alloy X-750 which is not a screened material.

Table 4 is a summary of nickel based alloy components in the reactor vessels contained in the Alloy 600 Program and supporting documents. It is recognized that this screening may not contain a complete list of nickel based components in the reactor vessels due to the scope limitations of the programs. This seems to create the possibility of 'orphan' components which may be covered only by the ASME B&PV Code.

The review of applicable programs shows that all accessible components in the reactor vessel are monitored by at least one existing program. It also revealed that some components are evaluated and monitored more rigorously based on function, safety significance, operational significance, material, susceptibility, and OE.

The EPRI created MRP-274, "Materials Reliability Program: Assessment of Westinghouse and Combustion Engineering Nickel-Based Alloy Orphan Locations," which discusses nickel-based orphan locations in the RCS. The discussion focuses on locations in the reactor vessel, fabricated primarily from Alloy 600 and 82/182, which have not been addressed in previous evaluations. Alloys 690 and 52/152 were excluded due to existing programs or studies. Alloy X-750 was excluded due to its limited and specific uses inside the reactor vessel. Upper and lower head penetrations, and inlet and outlet nozzles have been excluded due to existing programs or studies.

MRP-274 mentions applications of Alloy X-750 in Westinghouse plants including CRGT support pins (split pins) and clevis insert bolts. CRGT support pin OE was discussed and the LRSS clevis bolt condition observed at DC Cook Unit 1 was discussed. The report states the following:

> "Recent experience has confirmed that there is a need to inspect this component for potential aging degradation and has also confirmed that the ASME Code Section XI inspection is adequate to detect degradation of these bolts."

The list of austenitic nickel-based material orphan locations in Westinghouse plants from MRP-274 has been recreated in Table 5.

The components in Tables 3, 4, and 5 were compiled and reviewed with additional comments and recommendations in Table 6. The review found the following components which may be susceptible to PWSCC, and have component specific actions to consider:

- Remaining original Unit 1 LRSS clevis insert bolts
- Unit 2 clevis insert bolts
- LRSS lug welds
- BMI Nozzles and Welds

Several mitigating techniques were mentioned in the report. RCS zinc addition may be a generic method to inhibit PWSCC initiation and propagation in nickel-based alloys. Surface stress improvements are component specific methods in development or available including, but not limited to, peening and burnishing. Mitigation, repair, or replacement with improved materials is a common option as well.

A rigorous component review is being performed to support NRC commitments related to the use of MRP-227-A in the RVI AMP. This review will include a list of each component in the reactor vessel internals including material. The results of this review from contract 1500016-156 should be reviewed to determine if any additional actions need to be taken for reactor internals aging management.

Risk of Operating with Degraded Clevis Insert Hardware

This evaluation determined that there were no safety or operational impacts to D.C. Cook Unit 1 from the degraded clevis insert bolting. It has been documented that continued operation with degraded bolting has potentially significant risk involved. The risk of continued operation with degraded clevis insert hardware is described in Exhibit 25 as follows:

In the case of Unit 2, increased operation with degraded bolting can lead to increased wear on surrounding components. With increased operating times, it is possible that a clevis insert to clevis lug interference fit could be lost, and a new clevis insert would be required to be installed. This would require a significant repair operation that has never been attempted before. Even if the interference fit is not lost, continued operation with degraded bolting could lead to the clevis insert becoming loose and potentially damaging surrounding components. This could lead to future repairs being more extensive than would otherwise be necessary.

While the scenario discussed above would no longer apply to D.C. Cook Unit 1, the risk of loose parts migration must still be considered for any remaining original design bolting in either Unit 1 or 2. As operation continues with degraded clevis insert bolts, the probability that a loose part becomes small enough to enter the fuel assembly would increase. This is primarily because continued operation in the degraded condition will increase the potential for a larger number of loose parts. Additionally, the increased operating time will allow any loose parts that are created to tumble within the reactor internals and potentially break or wear into smaller parts. Thus, the loose parts, which in their nominal condition would typically be trapped by the fuel assembly debris filter bottom nozzles, could be reduced in size enough to pass the nozzle and enter the fuel assembly.

In addition to the risks of potential fuel leaks that continued operation presents, there are also risks associated with core reload activities. As previously discussed, the potential for loose parts to reduce in size and increase in quantity increases with continued operation. The smaller pieces and larger quantity of loose parts could make foreign object search and retrieval (FOSAR) activities more challenging. As a result, more care and time would be required to ensure that all parts are retrieved. Should pieces be missed during the FOSAR activities, fuel misalignment issues and additional steps to remove loose parts could result.

Each CNP unit has a Digital Metal Impact Monitoring System (DMIMS) to identify loose parts. The first place that loose parts may be detected is at the reactor vessel lower plenum where there are two sensors. The DMIMS did not detect the Unit 1 LRSS clevis bolt degradation with broken bolt heads vibrating within their installation counterbores. These sensors are calibrated to detect a loose part as small as 0.25 lbs, but the largest estimated loose part generated and released into the RCS flow from failed LRSS clevis bolts during operation is 0.039 lbs. Therefore, DMIMS is not expected to detect a degraded LRSS clevis bolt condition, or loose parts generated from such a condition during operation.

A more detailed evaluation on the risks associated with operating with degraded clevis insert hardware can be found in Exhibit 25.

See Sections 1.3 and 1.4 for actions generated as a result of this review.

Table 3: Summary of nickel based components considered in development of RVI AMP guidance documents.

				MRP-191[2]								
Assembly	**Sub-Assembly**	**Component**	**Material**	**None**	**SCC**	**Wear**	**Fatigue**	**IMT Conseq. Of Failure**	**Likelihood of Failure L, M, H**	**Likelihood of Damage L, M, H**	**Unit 1 Details/ Management**	**Unit 2 Details/ Management**
Upper Internals Assembly	Control Rod Guide Tube Assemblies and Flow Downcomers	Flexures	A X-750		SCC			G	H	M	Component is fabricated from 304 SST, screening from this line no applicable	Component is fabricated from 304 SST, screening from this line no applicable
		Guide Tube Support Pins	A X-750		SCC	Wear	Fat	NONE	H	M	GT-LRP 2012-1808, Replace in 2016	GT-LRP 2012-1809, Replace in 2016
		Support pin nuts	A X-750	X				NONE			GT-LRP 2012-1808, Replace in 2016	GT-LRP 2012-1809, Replace in 2016
Interfacing Components	Interfacing Components	Clevis insert bolts	A X-750		SCC	Wear		G	M	L	EC-51640, Replaced minimum pattern in 2010	
		Clevis insert lock keys (lock bars)	A 600	X				G			EC-51640, Partial removal in 2010	
		Clevis inserts	A 600			Wear		G	L	L		

G - Causes significant economic impact

Table 4: Summary of nickel based components inside the reactor vessel considered in the Alloy 600 Program

		Description			Inspection				Susceptibility					
Component	**Unit**	**Location**	**Material**	**Description**	**Method**	**Requirement**	**Frequency**	**Acceptance Criteria**	**Unit**	**Description**	**Alloy**	**Effective Stress (MPa)**	**Service Temp (F)**	**Susceptibility Index[3]**
Reactor Vessel	1, 2	Core Support Pads (LRSS Clevis Lugs and Insert)	Alloy 600 Alloy 82/182	The core support pads are fabricated from Alloy 600 and they are attached to the reactor vessel using Alloy 82/182 welds.	VT-3	ASME Section XI, Category B-N-2, item B13.60	Once per interval (per inspection requirement)	Per inspection requirement	1	Core Support Pad (at weld) (LRSS Lugs)	600	306.8	528.2	2.72E-11
										Core Support Pad Weld (LRSS Lug Welds)	82/182	348.1	528.2	2.71E-11
										Core Support Pad (LRSS Lugs)	600	29.2	528.2	2.24E-15
									2	Core Support Pad (at weld) (LRSS Lugs)	600	306.8	544	5.56E-11
										Core Support Pad Weld (LRSS Lug Welds)	82/182	348.1	544	5.53E-11
										Core Support Pad (LRSS Lugs)	600	29.2	544	4.56E-15
Reactor Vessel	1, 2	Reactor Vessel Internals	Alloy 600	The clevis inserts are fabricated from Alloy 600 material.	VT-3	ASME Section XI, Category B-N-2, item B13.60	Once per interval (per inspection requirement)	Per inspection requirement	WCAP-16198-P states "The clevis inserts (R 6) are bolted within the reactor vessel internals and are at approximately cold-leg temperatures. The only known incident where cracking of this component was suspected was subsequently found by destructive examination to be limited to cracking of the hard facing on the Alloy 600 insert and not cracking of the insert itself. Since this component does not support tensile loads and does not have a history of failure related to PWSCC, it was not assigned a susceptibility index."					
Reactor Vessel	1, 2	BMI Nozzles	Alloy 600 Alloy 82/182	The Alloy 600 nozzles are connected to the reactor vessel using Alloy 82/182 partial penetration welds. The welds connecting the BMI nozzles to the stainless steel guide tubes are also Alloy 82/182	Bare Metal Visual	ASME Code Case N-722-1, Item B15.80	Each refueling outage (only required every other refueling outage)	Per inspection requirement	1	BMI Nozzles	600	455.1	528.2	2.64E-10
										BMI Nozzle to Guide Tube Welds	82/182	393	528.2	4.39E-11
									2	BMI Nozzles	600	455.1	544	5.38E-10
										BMI Nozzle to Vessel Welds	82/182	410.9	544	1.07E-10
										BMI Nozzle to Guide Tube Welds	82/182	393	544	8.96E-11

[2] Columns with relevant information from MRP-191, Tables 4-4, 5-1, and 6-5, columns with no information for these components have been omitted.
[3] Susceptibility Index value is used to compare susceptibility of components, more detail can be found in the Alloy 600 Program and supporting documentation.

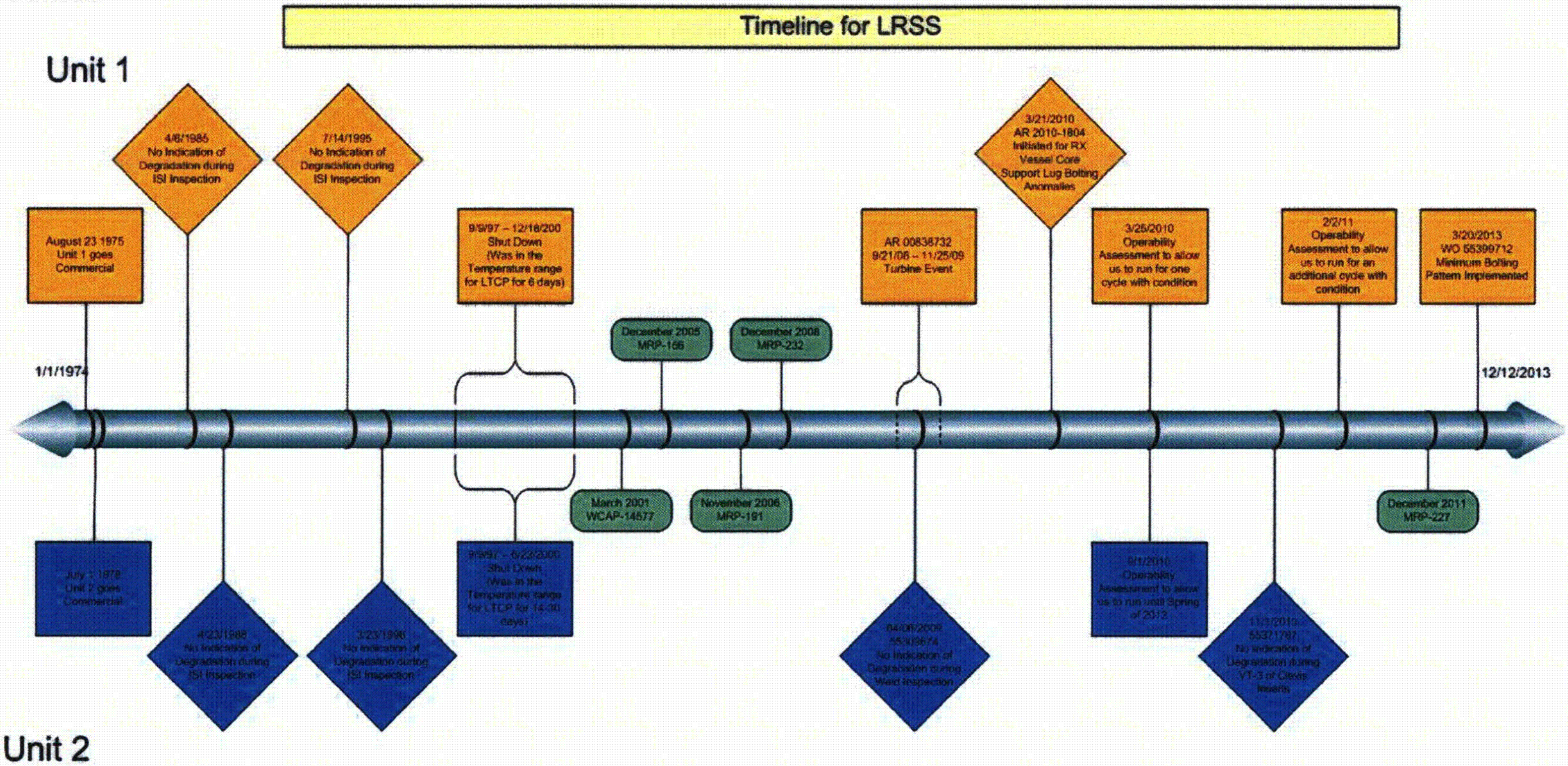
Table 5: List of Orphan Locations in Westinghouse-Designed PWRs from MRP-274

Component/Subcomponent	Material	Temperature
Core Support Lugs:		
Core support lugs	Alloy 600	Tcold
Core support lug weld	Alloy 182/82	Tcold
Clevis Inserts:		
Clevis Inserts	Alloy 600	Tcold
Leak-Off Monitor Tubes:		
Leak-off monitor tubes	Alloy 600	Mean head temp.
Leak-off monitor tube welds	Alloy 182/82	Mean head temp.

Table 6: Comments and Recommendations for Reviewed Components

Item	Material	Comments and Recommendations
Clevis Insert Bolts	Alloy X-750	See CATPR - section 1.3
Clevis Insert Bolt Lock Bars	A 600	Failure of this item is expected to be secondary to bolt failures and do not provide a structural function, therefore no additional actions recommended
Clevis Dowel Pins	A 600	Failure of this item is expected to be secondary to bolt failures. EHI-5070-ALLOY600 which provides a PWSCC susceptibility screening of alloy 600 components does not list clevis dowel pins, hence an action is recommended to update this procedure to include the dowel pins.
Clevis Inserts	A 600	Susceptibility of this component to PWSCC is low as observed in a basis document to the Alloy 600 Program [Exhibit 99], no known economically viable mitigation solution is available, therefore, no additional actions recommended
LRSS Lug	A 600	WCAP-16198-P states "Except for the location immediately adjacent to the weld, the residual stress in the core support pad [LRSS Lug] is negligible." Therefore, management of the lug would be bounded by management of the lug weld, no additional actions recommended
LRSS Lug Weld	82/182	Mitigating actions for this weld may include remote underwater peening, recommend investigating need and availability of options CA# 2010-1804-29
Guide Tube Support Pins	Alloy X-750	Current strategy of replacement is adequate, no additional actions recommended
Support Pin Nuts	Alloy X-750	Current strategy of replacement is adequate, no additional actions recommended
BMI Nozzles and Welds	A 600 and 82/182	Code case N-722-1 calls for a bare metal visual examination every other refueling outage. CNP performs this examination every outage to identify leakage. Leakage was identified on a BMI at Palo Verde in October of 2013. CNP is actively monitoring the industry response to the event at Palo Verde through participation in the EPRI MRP. Recommend CNP evaluate if potential mitigating actions exist to prevent PWSCC of the BMIs. CA# 2010-1804-30

5.3 EVENT TIMELINE



Industry Guidance Documents

WCAP-14577
Indicated susceptible material, however the effects of PWSCC of the clevis insert bolts are not significant and no failures have been observed (Page 103 & 104)

MRP-156
EPRI shows that failure of a clevis insert would cause a significant economic impact. Significant events are those for which we do not have a proven fix and would result in significant regulatory and/or scrutiny, such as-first-of-a-kind consideration would be suitable test. It can be considered that "non-significant" events are those for which it is expected that a proven fix exits that will require minimal regulatory and/or public scrutiny. (Pg. 324 & 397)

MRP-191
Alloy X-750 Clevis Insert Bolts have a medium likelihood of failure from PWSCC, but low consequence of failure. Also indicates Effective Stress>=Threshold for PWSCC, however Clevis Insert Bolts are not identified as Lead Components. Lead Components get further guidance in MRP-227. (Page 117 & 168)

MRP-232
Alloy X750 Clevis Insert Bolts re-categorized to category A in which aging degradation significance is minimal. (page 71 & 36)

MRP-227
Guidance is to do 10 year ISI visual

5.4 FAILURE MODES AND EFFECTS ANALYSIS

This chart was built from an equipment failure analysis technique. See Potential Failure Mode Evidence Matrix for details.

Unit 1 LRSS Clevis Bolt and Dowel Pin Failure							
1. Design Failure	**2. Material Selection Failure**	**3. Manufacturing or Fabrication Failure**	**4. Installation or Construction Failure**	**5. Operation Failure**	**6. Maintenance Failure**	**7. External Condition Induced Failure**	**8. Sabotage**
1.1. Overload Failure 1.1.1. Overstress 1.1.2. Bolting Sizing 1.1.3. High peak stress due to applied and residual stress **1.2. Low Cycle Fatigue (LCF)** 1.2.1. Thermal Cycling 1.2.2. Refuel Cycle 1.2.3. Flow Transients 1.2.4. Pump Issues 1.2.5. Pressure Overload 1.2.6. Irradiation Induced Loss of Preload **1.3. High Cycle Fatigue (HCF)** 1.3.1. Flow Induced Vibration 1.3.1.1. Pump Pulsation 1.3.1.2. Lower Internals Structure Vibration 1.3.1.3. Reactor Vessel Vibration 1.3.1.4. Vortex Shedding 1.3.2. Mechanically Induced Vibration 1.3.3. Electrically Induced Vibration (ex. Motor) 1.3.4. Magnetically Induced **1.4. Corrosion Failure** 1.4.1. Intergranular Stress Corrosion Cracking (IGSCC) 1.4.1.1. Primary Water SCC (PWSCC) 1.4.2. Irradiation Assisted SCC (IASCC) 1.4.3. Low Temperature Crack Propagation (LTCP) 1.4.4. Flow Assisted Corrosion **1.5. Erosion Failure** 1.5.1. Flow Turbulence 1.5.2. High Flow Velocity 1.5.3. Direct Flow Impingement **1.6. Inadequate Bolt Dimensions** **1.7. Inadequate Locking Mechanism** **1.8. Inadequate Torque Design Value** **1.9. Inadequate Tolerance Specifications** 1.9.1. Bolt head to shank perpendicularity out of square 1.9.2. Core Barrel Hang-Up or L/D Cocking 1.9.3. Clevis/Lug Misalignment 1.9.4. Counter bore Misalignment	**2.1. Bolt Selection of Alloy X-750** **2.2. Loading Strength Selection** **2.3. Fatigue Strength Selection** **2.4. Creep Strength Selection** **2.5. Embrittlement** 2.5.1. Irradiation Embrittlement 2.5.2. Thermal Embrittlement **2.6. Machinability** **2.7. Void Swelling** 2.7.1. Loss of Preload due to Void Swelling 2.7.2. Overload due to Void Swelling **2.8. Improper Material Heat Treatment Specification**	**3.1. Inadequate Clearance Specifications between Key and Clevis** **3.2. Inadequate Machining (Outside of drawings or specifications)** 3.2.1. Improper Surface Treatment (including after EDM) 3.2.2. Machine/Tool Marks Creat Notch/Defect/Stress Riser (Within drawing specification) **3.3. Inadequate Performance of Material Heat Treatment** 3.3.1. Excessive Residual stress in Clevis 3.3.2. Excessive Residual Stress in Lugs 3.3.3. Excessive Residual Stress in bolts **3.4. Inadequate Welding** 3.4.1. Inadequate Weld Preheat – Locking Bar 3.4.2. Inadequate PWHT – Locking Bar 3.4.3. Improper filler Metal – Locking Bar 3.4.4. Inadequate Weld Process Control – Locking Bar 3.4.5. Residual Stress in Lugs from Welding 3.4.6. Residual Stress in Clevis Insert from Welding 3.4.7. Distortion in Lugs from Welding 3.4.8. Distortion in Clevis Insert from Welding **3.5. Bottomed Out Bolting** **3.6. Improper Internal Thread Form** **3.7. Improper External Thread Form**	**4.1. Inadequate Torqueing of Bolts** 4.1.1. Over-Torqueing 4.1.2. Under-Torqueing 4.1.3. Inadequate Torque Pattern 4.1.4. Torqueing sequence when installing insert with respect to thermal expansion from low temperature exposure **4.2. Inadequate Orientation of Parts** **4.3. Inadequate Parts/Wrong Batch of Bolts Used** **4.4. Missing Parts** **4.5. Clevis Separation at Installation** **4.6. Clevis Deformation at Installation** **4.7. Lug Deformation at Installation** **4.8. Mechanical Scratch/Nick/Defect Caused During Assembly**	**5.1. Temperature Operating Condition out of Specification** **5.2. Mishandling of Internals Due to Human Errors** **5.3. Clevis Insert Distortion** **5.4. Thermal Expansion Differentials** **5.5. Overload on Defuel/Refuel** **5.6. Overload on Installation Lower Internals** **5.7. Overload on Removal of Lower Internals** **5.8. Overload Across Lower Internals at Start-Up of Individual Loop RCPs** **5.9. Mis-Operation Due to Failure of Another Component** 5.9.1. Failure in Thermal Shield 5.9.2. Failure in Core Barrel 5.9.3. Failure of Hold Down Spring 5.9.4. Flow Disturbances due to Dislodged Thermal Sleeves 5.9.5. Hold Down Spring Relaxation **5.10. Random Turbulence Excitation** **5.11. Thermally Induced Bolt Loss of Preload** **5.12. Irradiation Induced Bolt Loss of Preload** **5.13. Inadequate Water Chemistry with Respect to PWSCC** **5.14. Inadequate Water Chemistry with Respect to LTCP** **5.15. Foreign material Bound in Clevis/Key Clearance** **5.16. Configuration of Cold Leg Impingement Inducing Mechanical Loads** **5.17. Individual Coolant Loop Flow Imbalance**	**6.1. Preventive Maintenance** 6.1.1. Inadequate Inspection Technique 6.1.2. Inadequate Inspection Frequency 6.1.3. Failure to Apply OE **6.2. Corrective Maintenance** 6.2.1. Inadequate Repair of Barrel Former Bolts **6.3. Distortion from Hydraulic Testing**	**7.1. Earthquake** **7.2. 2008 Turbine Failure Excitation** 7.2.1. System Transients 7.2.2. Mechanical Excitation	**8.1 Inadequate Security and Surveillance**

5.5 SUPPORT REFUTE MATRIX

POTENTIAL FAILURE MODE EVIDENCE SUPPORT/REFUTE MATRIX			
POTENTIAL FAILURE MODE	**SUPPORTING INFORMATION**	**REFUTING INFORMATION**	**S/R[4]**
1. Design Failure			
1.1. Overload Failure 1.1.1. Overstress 1.1.2. Bolt Sizing 1.1.3. High peak stress due to applied and residual stress	• None	• No signs of overload at initiation sites of fracture surfaces as seen in Hot Cell Report. • Stresses met code allowables as seen in the Minimum Bolting Pattern Analysis • Per Design met codes allowables as seen in the Minimum Bolting Pattern Analysis. • No signs of overload at initiation location as seen in Hot Cell Report.	R
1.2. Low Cycle Fatigue (LCF) 1.2.1. Thermal Cycling 1.2.2. Refuel Cycle 1.2.3. Flow Transients 1.2.4. Pump Issues 1.2.5. Pressure Overload 1.2.6. Irradiation Induced Loss of Preload	• None	• No signs of Low Cycle Fatigue on fracture surfaces as seen in Hot Cell Report.	R
1.3. High Cycle Fatigue (HCF) 1.3.1. Flow Induced Vibration 1.3.1.1. Pump Pulsation 1.3.1.2. Lower Internals Structure Vibration 1.3.1.3. Reactor Vessel Vibration 1.3.1.4. Vortex Shedding 1.3.2. Mechanically Induced Vibration 1.3.3. Electrically Induced (ex. Motor) 1.3.4. Magnetically Induced	• None	• No signs of High Cycle Fatigue on fracture surfaces as seen in Hot Cell Report.	R

[4] Support (S), Refute (R), Root Cause (RC), Contributing Cause (C), Potential Contributing Cause (PC)

POTENTIAL FAILURE MODE EVIDENCE SUPPORT/REFUTE MATRIX			
POTENTIAL FAILURE MODE	**SUPPORTING INFORMATION**	**REFUTING INFORMATION**	**S/R**
1.4. Corrosion Failure 1.4.1. Intergranular Stress Corrosion Cracking (IGSCC) 1.4.1.1. Primary Water SCC (PWSCC)	**IGSCC/PWSCC** • CN-RIDA-10-27 Appendix A shows localized peak stresses due to stress risers at the head to shank radius far greater then yield stress and recommended stresses to stay below to prevent SCC. • Literature shows Alloy X-750 solution annealed below 1800 degrees F are susceptible to SCC in a PWR environment [Exhibit 12] • Fracture faces showed signs of IGSCC at initiation site and for nearly full propagation distance (some overload at final failure). • Local Environment for bolts was Primary Water (PW)	**IGSCC/PWSCC** • None	**IGSCC/PWSCC** S-RC
1.4.2. Irradiation Assisted SCC (IASCC)	**IASCC** • CN-RIDA-10-27 Appendix A shows localized peak stresses far greater then yield stress and recommended stresses to stay below to prevent SCC • Literature shows Alloy X-750 solution annealed below 1800 degrees F are susceptible to SCC.	**IASCC** • Not in an area where you would expect IASCC • Fluence level significantly lower than threshold for IASCC	**IASCC** R
1.4.3. Low Temperature Crack Propagation (LTCP)	**LTCP** • LTCP can cause existing cracks and sharp surface defects to grow at a rapid rate.	**LTCP** • LTCP may have made cracks propagate faster, but did not initiate them. • 12-thp-6020-chm-100 fig. 1 shows RCS dissolved Hydrogen target bands met recommendations for prevent LTCP starting in 2002 • **Note: These refute LTCP from being a root cause, but these do not refute LTCP from being a potential contributing cause	**LTCP** S-PC
1.4.4. Flow Assisted Corrosion	**Flow Assisted Corrosion** • None	**Flow Assisted Corrosion** Initial Fracture is in a non-flow area	**Flow Assisted Corrosion** R
1.5. Erosion Failure 1.5.1. Flow Turbulence 1.5.2. High Flow Velocity 1.5.3. Direct Flow Impingement	• Area of high flow	• Initial Fracture is in a non-flow area • Location of cracks would not have seen flow impingement • No signs of erosion in metallurgical report.	R

POTENTIAL FAILURE MODE EVIDENCE SUPPORT/REFUTE MATRIX			
POTENTIAL FAILURE MODE	**SUPPORTING INFORMATION**	**REFUTING INFORMATION**	**S/R**
1.6. Inadequate Bolt Dimensions	• Radius causes a stress riser that can be reduced by design but not be designed out. Note: The stresses produced at the head to shank radius is not sufficient to cause overload failure on their own, but this was found to be a contributing cause to PWSCC as shown in line 1.4.1.	• None for bolt head to shank transition • Remaining bolt dimensions are adequate to ensure proper bolt strength.	S-C
1.7. Inadequate Locking Mechanism	• None	• No Locking Bar failures	R
1.8. Inadequate Torque Design value	• None	• Minimum Bolt Pattern supported same preload	R
1.9. Inadequate Tolerance Specifications 1.9.1. Bolt head to shank perpendicularity out of square 1.9.2. Core Barrel Hang-Up or L/D Cocking 1.9.3. Clevis/Lug Misalignment 1.9.4. Counter bore Misalignment	• All bolts were cracked therefore measurements could not be accurately taken. • Bolt crack orientations show no particular pattern which suggests that if there was a dimensional defect causing an A-symmetric load that it would be on the bolts, not on other joint features.	• No inadequate tolerances seen in Hot Cell report. • Core Barrel Hang up would only be a concern for overload • All 29 bolts failed • We saw no overload that would have yielded the bolt. • The method of manufacturing bolts machines most external features in one setup on a lathe making misalignment difficult. • No signs of overload • Pictures from bolt replacement show concentricity between clevis holes and lug holes • No pattern to the orientation of fracture surface lines of symmetry. • This feature of the clevis inserts was manufactured in a shop and therefore would expect to have a consistent effect on the symmetry of fracture surfaces.	R
2. Material Selection Failure			
2.1. Bolt Selection of Alloy X-750	• None	• Replacement material had same chemistry to support thermal expansion coefficient and strength considerations • Alloy X-750 was/is considered the proper bolting material because of strength and general corrosion properties	R
2.2. Loading Strength Selection	• None	• Alloy X-750 typically selected for high strength properties	R
2.3. Fatigue Strength Selection	• None	• No signs of fatigue on fracture surfaces	R
2.4. Creep Strength Selection	• None	• Temperature is too low. • No evidence of creep in Hot Cell Report.	R
2.5. Embrittlement 2.5.1. Irradiation Embrittlement 2.5.2. Thermal Embrittlement	• None	• Neutron fluence is 2.2X10^-3 DPA at the clevis insert bolts, this is below the threshold of 5 DPA for void swelling • Alloy X-750 not susceptible to Thermal Embrittlement	R

POTENTIAL FAILURE MODE EVIDENCE SUPPORT/REFUTE MATRIX			
POTENTIAL FAILURE MODE	**SUPPORTING INFORMATION**	**REFUTING INFORMATION**	**S/R**
2.6. Machinability	• None	• Machinable	R
2.7. Void Swelling 2.7.1. Loss of Preload due to Void Swelling 2.7.2. Overload due to Void Swelling	• None	• Neutron fluence is 2.2X10^-3 DPA at the clevis insert bolts, this is below the threshold of 5 DPA for void swelling • No evidence of overload in the initiation sites of fracture faces in Hot Cell Report.	R
2.8. Improper Material Heat Treatment Specification	• Literature shows Alloy X-750 solution annealed below 1800 degrees F are susceptible to SCC in a PWR environment. Note: This support was found to be a contributor to PWSCC as shown in line 1.4.1.	• None	S-RC
3. Manufacturing or Fabrication Failure			
3.1. Inadequate Clearance Specifications between Key and Clevis	• None	• No evidence of overload on bolting	R
3.2. Inadequate Machining (Outside of drawings or specifications) 3.2.1. Improper Surface Treatment (including after EDM) 3.2.2. Machine/Tool Marks Creat Notch/Defect/Stress Riser (Within drawing specification)	• None	• Hot Cell Report shows no evidence of improper surface treatment • Hot Cell report has been updated to show the radius is within specification. • The bolt head-to-shank interface was within size tolerance, but did not follow the intended surface profile. However, fractures were not observed to initiate at any suspect areas.	R
3.3. Inadequate Performance of Material Heat Treatment 3.3.1. Excessive Residual stress in Clevis 3.3.2. Excessive Residual Stress in Lugs 3.3.3. Excessive Residual Stress in bolts	• None	• No signs of cold working in the microstructure of the head to shank radius as seen in Hot Cell Report. • Residual stress is only a concern for the part in which the residual stress is in. • Test showed material behaved per specification.	R
3.4. Inadequate Welding 3.4.1. Inadequate Weld Preheat – Locking Bar 3.4.2. Inadequate PWHT – Locking Bar 3.4.3. Improper filler Metal – Locking Bar 3.4.4. Inadequate Weld Process Control – Locking Bar 3.4.5. Residual Stress in Lugs from Welding 3.4.6. Residual Stress in Clevis Insert from Welding 3.4.7. Distortion in Lugs from Welding 3.4.8. Distortion in Clevis Insert from Welding	• None	• Not a structural weld • No preheat • No PWHT • Residual stress is only a concern for the part in which the residual stress is in. • No welding on clevis insert beyond locking bar. Tacking welding of lock bar does not create enough heat to induce distortion. • Lock bar weld is a sufficient distance from hole and seating surface that there is no concern. • Holes in the lugs were field drilled following welding of the lugs, hence if distortion in the lugs existed this would not have an effect on the alignment of holes between the clevis and lugs.	R

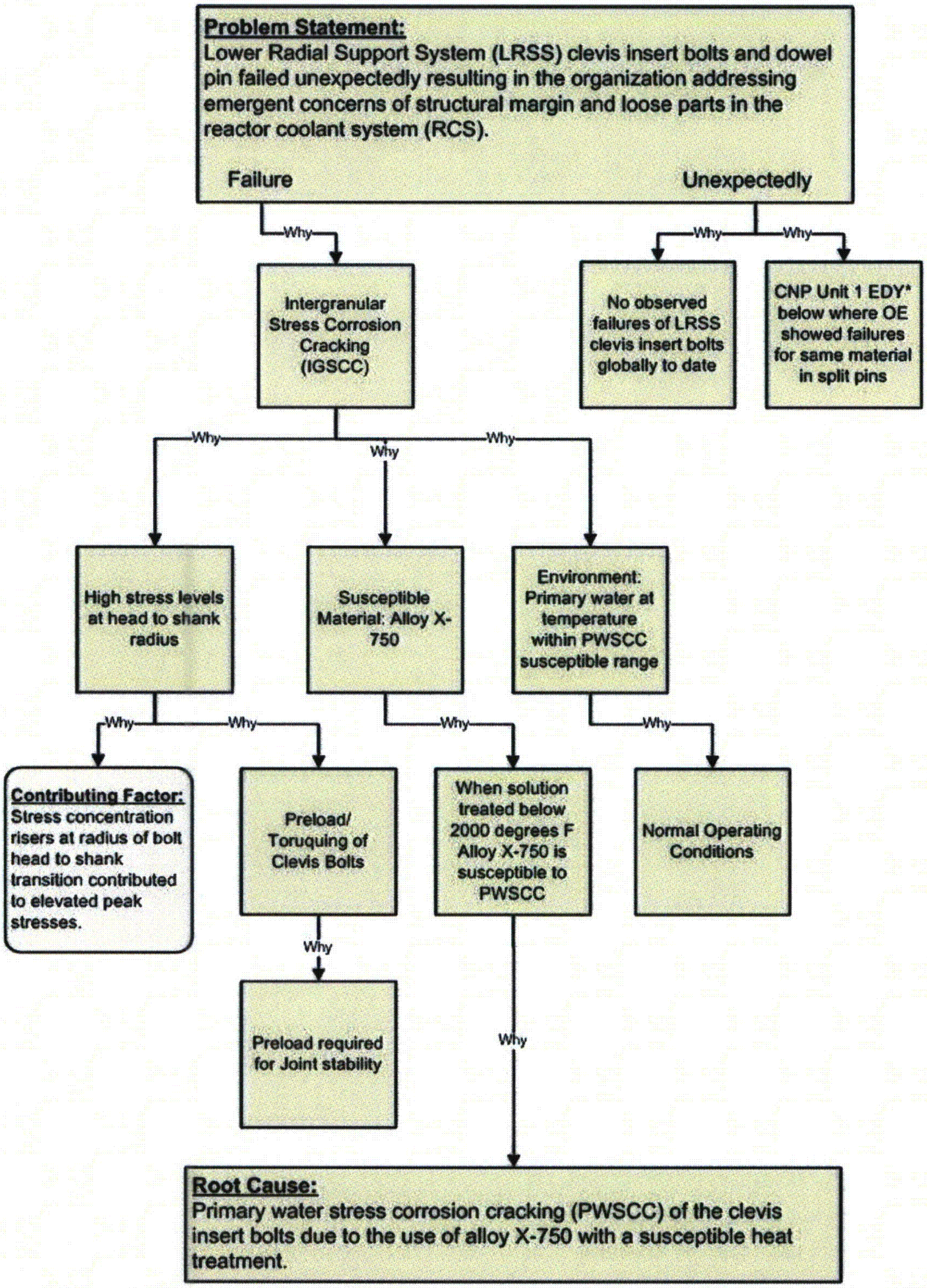
POTENTIAL FAILURE MODE EVIDENCE SUPPORT/REFUTE MATRIX			
POTENTIAL FAILURE MODE	**SUPPORTING INFORMATION**	**REFUTING INFORMATION**	**S/R**
3.5. Bottomed Out Bolting	• None	• Each bolt hole was inspected during bolt replacement campaign, all bolt holes had adequate depth.	R
3.6. Improper Internal Thread Form	• Replacement bolts experience galling at external thread crest during installation, major diameter turned down to LMC on remaining replacement bolts which solved problem	• Improper thread form would lead to failures in threaded region not the under head radius of bolt. • Galling in threaded region would reduce stresses at the head to shank radius if anything. • No indication of failure in the threads as indicated by the Hot Cell Report.	R
3.7. Improper External Thread Form	• None	• No indication of failure in the threads as indicated by the Hot Cell Report.	R
4. Installation or Construction Failure			
4.1. Inadequate Torqueing of Bolts 4.1.1. Over-Torqueing 4.1.2. Under-Torqueing 4.1.3. Inadequate Torque Pattern 4.1.4. Torqueing sequence when installing insert with respect to thermal expansion from low temperature exposure	• Insert was immersed in liquid nitrogen for shrinking, clevis bolts were torqued to 270-290 ft. lbs., the clevis warmed to room temperature, the bolts torqued to 555-575 ft. lbs, backed off and torqued to 555 ft lbs.	• All bolts are behaving the same so torque was consistent and following quality inspection procedure • Under torqueing would result in fatigue which was not evident in the Hot Cell Report. • No yielding of the bolt or overload seen at the initiation site of the fracture faces as seen in Hot Cell Report. • OEM implemented a rigorous quality program during construction to ensure critical parameters were met. • Quality review trip reports were provided which showed that deviations were documented and dispositioned appropriately.	R
4.2. Inadequate Orientation of Parts	• None	• Parts appeared to be in the correct orientation	R
4.3. Inadequate Parts/Wrong Batch of Bolts Used	• None	• Metallurgy matched that expected for heat treat	R
4.4. Missing Parts	• None	• No parts were missing	R
4.5. Clevis Separation at Installation	• None	• The Minimum Bolt Pattern Analysis analyzed the pre load on the bolts and determined the clevis would be pulled up to lug eliminating gap	R
4.6. Clevis Deformation at Installation	• Liquid nitrogen bath to shrink clevis insert for installation, lock bar tack welding	• If this was a dimensional installation issue then highly unlikely we would be seeing a generic bolt failure issues throughout all clevises.	R
4.7. Lug Deformation at Installation	• Lugs are welded to the RX wall	• If this was a dimensional installation issue then highly unlikely we would be seeing a generic bolt failure issues throughout all clevises.	R
4.8. Mechanical Scratch/Nick/Defect Caused During Assembly	• None	• The bolt head-to-shank interface was within size tolerance, but did not follow the intended surface profile. However, fractures were not observed to initiate at any suspect areas. • No cold working was observed on any bolts or microstructures.	R

POTENTIAL FAILURE MODE EVIDENCE SUPPORT/REFUTE MATRIX			
POTENTIAL FAILURE MODE	**SUPPORTING INFORMATION**	**REFUTING INFORMATION**	**S/R**
5. Operation Failures			
5.1. Temperature Operating Condition out of Specification	• None	• No discrepancies found for cold leg temperature via R time review.	R
5.2. Mishandling of Internals Due to Human Errors	• None	• Installation of the lower internals is performed in accordance with 12-OHP-4050-FHP-045. The internals lift rig engagement screws, guide bushings, reactor vessel guide studs, vessel alignment keys, lower radial support system keys and clevis inserts all work together to ensure alignment and proper installation of the lower internals into the reactor vessel. • No signs of overload at initiation sites of fracture surfaces as seen in Hot Cell Report.	R
5.3. Clevis Insert Distortion	• None	• Temperature and Operational loads are not sufficient to distort the clevis	R
5.4. Thermal Expansion Differentials	• None	• Thermal expansion differential were analyzed in the MBPA and was found to be adequate	R
5.5. Overload on Defuel/Refuel	• None	• No signs of overload on fracture faces	R
5.6. Overload on Installation Lower Internals	• Procedure allows installation of lower internals w/o Microdrive with plant manager approval	• No signs of overload on fracture faces • Installation of the lower internals is performed in accordance with 12-OHP-4050-FHP-045. The internals lift rig engagement screws, guide bushings, reactor vessel guide studs, vessel alignment keys, lower radial support system keys and clevis inserts all work together to ensure alignment and proper installation of the lower internals into the reactor vessel.	R
5.7. Overload on Removal of Lower Internals	• None	• No signs of overload on fracture faces • Removal of the lower internals is performed IAW 12-OHP-4050-FHP-044. To prevent misalignment during disassembly, the difference between internals lift rig attachment points is limited to one or less turns of the engagement screw, which is 1/4 inch or less of misalignment. For the Unit 1 lower internals removal in 2010, the maximum difference between screw engagement was recorded to be 1/4 turn (10 turns - 9 and 3/4 turns) which is 1/16 of an inch. [Exhibit 108].	R
5.8. Overload Across Lower Internals at Start-Up of Individual Loop RCPs	• None	• No signs of overload on fracture faces	R

POTENTIAL FAILURE MODE EVIDENCE SUPPORT/REFUTE MATRIX			
POTENTIAL FAILURE MODE	**SUPPORTING INFORMATION**	**REFUTING INFORMATION**	**S/R**
5.9. Mis-Operating Due to Failure of Another Component 5.9.1. Failure in Thermal Shield 5.9.2. Failure in Core Barrel 5.9.3. Failure of Hold Down Spring 5.9.4. Flow Disturbances due to Dislodged Thermal Sleeves 5.9.5. Hold Down Spring Relaxation	• U1 has a 304 SS hold-down spring which is susceptible to thermal "ratcheting", leading to permanent deformation which leads to reduced hold down force over time. This has a moderate likelihood of occurrence	• Hot Cell Report indicated an aging degradation mechanism • No related link between the 3 Former Bolts and Clevis • Per Roy Hall's Inspections • One indication during 1985 inspection - Indication appeared to be a tooling gouge, no other indication • Thermal sleeve OE evaluation performed by K. Kalchik in Away	R
5.10. Random Turbulence Excitation	• None	• Turbulence and flow induced vibration was analyzed in the Minimum Bolting Pattern and found acceptable	R
5.11. Thermally Induced Bolt Loss of Preload	• None	• Bolt preload at the as-installed and hot preload is described in the Minimum Bolting Pattern Analysis at the original design conditions and found to be adequate	R
5.12. Irradiation Induced Bolt Loss of Preload	• None	• Loss of preload is minimal and it does not have significant adverse effects as described in the Minimum Bolting Pattern Analysis	R
5.13. Inadequate Water Chemistry with Respect to PWSCC	• None	• LER and A-Way searches for chlorides, fluorides, and sulfates returned no relevant results.	R
5.14. Inadequate Water Chemistry with Respect to LTCP	• See 1.4.4	• See 1.4.4	
5.15. Foreign Material Bound in Clevis/Key Clearance	• None	• No indication of Foreign Material found there	R
5.16. Configuration of Cold Leg Impingement Inducing Mechanical Loads	• None	• No evidence of overload fatigue	R
5.17. Individual Coolant Loop Flow Imbalance	• None	• No A-symmetric failure identified	R
6. Maintenance Failure			

POTENTIAL FAILURE MODE EVIDENCE SUPPORT/REFUTE MATRIX			
POTENTIAL FAILURE MODE	**SUPPORTING INFORMATION**	**REFUTING INFORMATION**	**S/R**
6.1. Preventive Maintenance 6.1.1. Inadequate Inspection Technique 6.1.2. Inadequate Inspection Frequency 6.1.3. Failure to Apply OE	• Currently Inspections are only visual and this type of inspection will not find a failure in a bolt unless the head is broke off. • Significant industry experience with Alloy X-750 failure in RVI components such as split pins (support pins)	• Inspection would only tell you current condition of bolts • A different inspection technique would not decrease the probability of failure and no consequence of LRSS bolt failures on clevis system design function occurred • No Catastrophic failure • No other plants have experienced LRSS clevis bolt failures • Industry guidance indicates that bolt failures do not result in an immediate operability or safety concern. • CNP followed industry guidance which was approved by the NRC to perform VT-3 of clevis insert bolts.	R
6.2. Corrective Maintenance 6.2.1. Inadequate Repair of Barrel Former Bolts	• Method of Former Bolt failure inadequate to determine all failures • No clear way to determine bolt failure	• No Maintenance has ever been done on Clevis Insert bolts • A-symmetric failure for former bolts and symmetric failure for clevis bolts • No related link between the 3 Former Bolts and Clevis	R
6.3. Distortion from Hydraulic Testing	• None	• No deformation or overload	R
7. External Condition Induced Failure			
7.1. Earthquake **7.2. 2008 Turbine Failure Excitation** 7.2.1. System Transients 7.2.2. Mechanical Excitation	• None	• Cook has had 3 small earthquakes on record. All had a severity considerably less than the design OBE • The Unit 1 turbine failure caused a transient that would be similar to an earthquake, but the loads would have been considerably less than the design OBE • A missile block was dropped during a Unit 2 refueling outage, but this would not have caused adverse effects on reactor vessel internals based on information found in AR 00124109 • No large-scale flooding of equipment has occurred at Cook U1	R
8. Sabotage			
8.1. Inadequate Security and Surveillance	• None	• No access to component • Bolts were manufactured to Westinghouse specifications which was confirmed by Hot Cell	R

5.6 WHY STAIRCASE



* EDY - Effective Degradation Years is the calculated value to normalize operational time and temperature for expected degradation comparison of components between tests and plant data.

5.7 INDUSTRY GUIDANCE REVIEW FOR CLEVIS INSERT BOLTS

A summary of information gained from a review of industry guidance documents on the aging management of reactor vessel internals components with respect to clevis insert bolts is provided below. The EPRI MRP (Materials Reliability Program) documents were developed in a hierarchy fashion building off of each other to obtain a final inspection and evaluation guidelines for managing long-term aging of reactor vessel internal components of pressurized water reactors as documented in MRP-227-A. It should be noted that beyond the summary statement for each document, specifics provided below are based off statements in each document related to Alloy X-750 clevis insert bolts only.

1. March 2001: WCAP-14577, "License Renewal Evaluation: Aging Management for Reactor Internals"

   a. Summary: Evaluates aging of the reactor internals components to ensure that intended functions will be maintained during an extended period of operation. Endorsed by the NRC.

   b. Heat treatment employed could result in a material susceptible to PWSCC.

   c. Fluence, temperature, and stresses are lower for clevis insert bolts than support pins

   d. No clevis bolt degradation or cracking reported in any Westinghouse plants to date

   e. Beyond bolt failures additional components would have to fail to compromise clevis integrity

   f. Effects of PWSCC of the clevis insert bolts are not significant

   g. Credits continuation of the ISI VT-3 inspection

   h. Degradation can be detected before function is compromised – pg 129

2. December 2005: MRP (EPRI Materials Reliability Program)-156, "Pressurized Water Reactor Issue Management Table, PWR-IMT Consequence of Failure"

   a. Summary: Provides initial input to address the consequences of failure for the identified components in reactor coolant systems for operating US PWRs designed by Babcock & Wilcox, Combustion Engineering, and Westinghouse.

   b. Consequence of failure for clevis insert bolts and lock keys were identified as level G likely due to potential effect on clevis insert.

   c. Level G: Causes a significant economic impact. Significant events are those for which we do not have a proven fix and would result in significant regulatory and/or public scrutiny, such as first-of a-kind consideration would be a suitable test. It can be considered that "non–significant" events are those for which it is expected that a proven fix exists that will require minimal regulatory and/or public scrutiny.

3. November 2006: MRP-191, "Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design

   a. Summary: Screened, categorized, and ranked Westinghouse and CE designed PWR internals components based on probability of occurrence of a degradation mechanism, probability of failure, and the severity of any consequences with respect to operation, safety, financial impact and plant reliability with respect to 60 years of operation.

   b. Provided FMECA (Failure modes, effects, and criticality analysis) rankings based on failure likelihood of occurrence and consequence of failure with respect to safety, reliability, and economic risk. Clevis insert bolts received a 1 out of a 1-3 ranking with one being the lowest.

   c. Provided an initial categorization of components to either category A, B, or C:

      i. Category A: Component items for which aging effects are below the screening criteria. Additional components may be ultimately categorized as A as discussed in B below.

      ii. Category B: Defined as those component items that are above screening levels but not 'lead' components. Aging degradation significance is moderate. May require additional evaluations to be shown tolerant of the aging effects with no loss of functionality. If it is further concluded that the existing 10 year in-service inspection or other in-place aging management plans are sufficient to preclude a safety, reliability, or financial concern, such components can be reassigned as Category A

         1. Clevis insert bolts initially categorized as B based on SCC and Wear.

      iii. Category C: 'Lead' component items for which aging effects are above screening levels. Aging degradation significance is high or moderate. Enhanced/augmented inspections and/or surveillance sampling typically may be warranted to asses aging affects and verify component item functionality.

   d. Components that were a FMECA group 1 were binned into either A or B components. Components with a moderate probability of failure were placed in Category B. (page 131)

   e. Consequence of Failure: Category G = Significant economic impact

4. December 2008: MRP-232, "Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals" [Exhibit 88]

a. Summary: Summarizes the aging management strategy development for Westinghouse and Combustion Engineering (CE) reactor internals. Provides the technical basis for the aging management requirements of Westinghouse and CE reactor internals in MRP-227.

b. Final disposition on LRSS clevis insert bolt was a re-categorization from B to A which required a determination that there was 'no credible damage issue' for the component (page 36). Reclassification appeared to be based on low consequence of failure, not based on probability of stress corrosion cracking.

c. All B and C components then defined into a program group. Clevis insert bolts were binned into existing programs in which generic or plant-specific programs are capable of managing aging effects. Clevis insert bolts are currently inspected as part of the ten year ISI inspection.

5. December 2011: MRP 227-A, "Pressurized Water Reactor Internals Inspection and Evaluation Guidelines"

   a. Summary: Provides inspection and evaluation guidelines for managing long-term aging reactor vessel internal components of pressurized water reactors reactor internals. Endorsed by the NRC.

   b. Reiterates category A for stress corrosion cracking and management by existing programs.

   c. Recommends inspections of the clevis insert for loss of material or wear per the ASME Code Section XI, In Service Inspection utilizing a visual (VT-3) examination looking at all accessible surfaces at specified frequency.

5.8 ORGANIZATIONAL AND PROGRAMMATIC FAILURE MODE REVIEW

Organizational & Programmatic Failure Modes Review					
O&P Failure Mode	**Failure Mode**	**Failure Mode Definition**	**Did the evaluation identify any Organizational & Programmatic Failure Modes? Yes/No (Y/N)**		**Applicable Corrective Actions**
			Y/N	**Basis**	
Organizational Deficiencies (Culture)	Organizational Structure (O1)	Inadequate organizational structure, span of control, or levels within the organization.	No	The issues reviewed in this evaluation did not provide any evidence regarding inadequate organizational structure, span of control, or levels within the organization.	N/A
	Organizational Teamwork (O6)	Inadequate teamwork or communication within the organization.	No	The issues reviewed in this evaluation did not provide any evidence regarding inadequate teamwork or communication within the organization.	N/A
	Rewards / Sanctions (O7)	Inappropriate rewards or sanctions.	No	No rewards or sanctions were reviewed as a part of this evaluation.	N/A
	Accountability (O8)	Lack of commitment or accountability.	No	The issues reviewed in this report found no lack of commitment or accountability.	N/A
	Engagement (O9)	Lack of employee engagement.	No	The issues reviewed in this report found no lack of employee engagement.	N/A
	Empowerment (O10)	Lack of employee empowerment.	No	The issues reviewed in this report found no lack of employee empowerment.	N/A
	Employee Development (O11)	Lack of employee development.	No	The issues reviewed in this report found no lack of employee development.	N/A
	Knowledge Management (O12)	Knowledge management weaknesses; including insufficient management of key organization knowledge or skills.	No	The issues reviewed in this report found evidence of knowledge management weaknesses. All evidence shows that the clevis insert bolts were known to have a potential for failure, but because of the low significance of failure and no history of failure in the industry, no contingencies were in place in the event of a failure. This was the industry standard.	N/A
	Staffing (OP3a)	Insufficient staffing or resources.	No	The issues reviewed in this report found no evidence of insufficient staffing or resources.	N/A
	Decision Making (O5a)	Weak organizational decision making.	No	The issues reviewed in this report found no evidence of weak organizational decision making.	N/A

Organizational & Programmatic Failure Modes Review					
O&P Failure Mode	**Failure Mode**	**Failure Mode Definition**	**Did the evaluation identify any Organizational & Programmatic Failure Modes? Yes/No (Y/N)**		**Applicable Corrective Actions**
			Y/N	**Basis**	
Programmatic Deficiencies	Program Definition (P2)	Insufficient program definition or scope.	No	The issues reviewed in this report found no evidence of insufficient program definition or scope.	N/A
	Program Requirements (P3)	Insufficient, excessive, or conflicting program requirements.	No	The issues reviewed in this report found no evidence of insufficient, excessive, or conflicting program requirements.	N/A
	Program Commitment (OP1)	Lack of commitment to program implementation.	No	The issues reviewed in this report found no evidence of a lack of commitment to program implementation.	N/A
	Program Monitoring (OP2)	Inadequate program monitoring, evaluation, or management.	No	The issues reviewed in this evaluation supported the 10 year ISI VT-3 examination as the appropriate periodicity and inspection based on the low safety significance of a clevis bolt failure. WCAP-14577 Rev 1-A, March 2001, states, "the effects of PWSCC of the clevis insert bolts are not significant". The recently revised (since the D.C. Cook clevis insert bolt failure) MRP-227-A, MRP-227 Roadmap, shows the clevis insert bolts are considered to have a low likelihood of damage, medium likelihood of failure, and a consequence of failure being only a significant economic impact.	N/A
	Program to Program Interface (PP4)	Weaknesses or lack of interface between programs.	No	The issues reviewed in this report found no evidence of weaknesses or lack of interface between programs.	N/A
	Organization to Program Interface (OP4)	Lack of organizational authority or engagement for program implementation.	No	The issues reviewed in this report found no evidence of lack of organizational authority or engagement for program implementation.	N/A
	Organization to Organization Interface (OO3)	Lack of interface between two organizational groups for a program.	No	The issues reviewed in this report found no evidence of a lack of interface between two organization groups for a program.	N/A

Organizational & Programmatic Failure Modes Review					
O&P Failure Mode	**Failure Mode**	**Failure Mode Definition**	**Did the evaluation identify any Organizational & Programmatic Failure Modes? Yes/No (Y/N)**		**Applicable Corrective Actions**
			Y/N	**Basis**	
	Change Management (J)	Weaknesses in implementation of people, processes, or equipment changes.	No	The issues reviewed in this report found no weaknesses in implementation of people, processes, or equipment changes.	N/A

5.9 SAFETY CULTURE IMPACT REVIEW

Safety Culture Impact Review					
NRC Components of Safety Culture					
Safety Culture: Cross-Cutting Area	**Safety Culture: Cross-Cutting Area Component**	**Safety Culture: Cross-Cutting Area Component Definition**	**Did the evaluation identify any of the following Cross-Cutting Area Components? Yes/No (Y/N) Document the basis for the safety culture Cross-Cutting Area Component as related or not.**		**Reference corrective actions addressing any safety culture Cross-Cutting Area Component noted as yes.**
			Y/N	**Basis**	
Human Performance	Decision Making	CNP decisions demonstrate that nuclear safety is an overriding priority.	No	This evaluation found no crosscutting issues that affected nuclear safety.	N/A
	Resources	CNP ensures that personnel, equipment, procedures, and other resources are available and adequate to assure nuclear safety.	No	The evaluation did not identify any challenges to nuclear safety. No personnel, equipment, procedure, or other resource issues that would impact nuclear safety were identified by this evaluation.	N/A
	Work Control	CNP plans and coordinates work activities, consistent with nuclear safety.	No	No work control issues were identified in this evaluation that had any impact on nuclear safety.	N/A
	Work Practices	Personnel work practices support human performance.	No	No work practice issues were identified by this evaluation that had any impact on nuclear safety.	N/A

Safety Culture Impact Review					
NRC Components of Safety Culture					
Safety Culture: Cross-Cutting Area	**Safety Culture: Cross-Cutting Area Component**	**Safety Culture: Cross-Cutting Area Component Definition**	**Did the evaluation identify any of the following Cross-Cutting Area Components? Yes/No (Y/N) Document the basis for the safety culture Cross-Cutting Area Component as related or not.**		**Reference corrective actions addressing any safety culture Cross-Cutting Area Component noted as yes.**
			Y/N	**Basis**	
Problem Identification and Resolution	Corrective Action Program	CNP ensures that issues potentially impacting nuclear safety are promptly identified, fully evaluated, and that actions are taken to address safety issues in a timely manner, commensurate with their significance.	No	No issues were identified by this evaluation regarding inadequate us of the Corrective Action Program.	N/A
	Operating Experience	CNP uses OE information, including vendor recommendations and internally generated lessons learned, to support plant safety.	No	No issues were identified by this evaluation regarding the use of Operating Experience.	N/A
	Self & Independent Assessment	CNP conducts self- and independent assessments of their activities and practices, as appropriate, to assess performance and identify areas for improvement.	No	No issues were identified by this evaluation regarding self and independent assessments.	N/A

Safety Culture Impact Review					
NRC Components of Safety Culture					
Safety Culture: Cross-Cutting Area	Safety Culture: Cross-Cutting Area Component	Safety Culture: Cross-Cutting Area Component Definition	Did the evaluation identify any of the following Cross-Cutting Area Components? Yes/No (Y/N) Document the basis for the safety culture Cross-Cutting Area Component as related or not.		Reference corrective actions addressing any safety culture Cross-Cutting Area Component noted as yes.
			Y/N	Basis	
Safety Conscious Work Environment	Environment for Raising Concerns	An environment exists in which employees feel free to raise concerns both to their management and/or the NRC without fear of retaliation, and employees are encouraged to raise such concerns.	No	This evaluation found no crosscutting issues that affected nuclear safety. There were no indications that employees have any concerns with reporting problems or raising concerns.	N/A
	Preventing, Detecting, and Mitigating Perceptions of Retaliations	A policy for prohibiting harassment and retaliation for raising nuclear safety concerns exists and is consistently enforced.	No	This evaluation found no crosscutting issues that affected nuclear safety. The existing policy prohibiting harassment and retaliation for raising nuclear concerns continues to exist.	N/A
Other	Accountability	CNP Management defines the line authority and responsibility for nuclear safety	No	This evaluation found no crosscutting issues with nuclear safety accountability.	N/A
	Continuous Learning Environment	CNP ensures that a learning environment exists.	No	This evaluation found no crosscutting issues with CNP ensuring a continuous learning environment.	N/A

Safety Culture Impact Review					
NRC Components of Safety Culture					
Safety Culture: Cross-Cutting Area	Safety Culture: Cross-Cutting Area Component	Safety Culture: Cross-Cutting Area Component Definition	Did the evaluation identify any of the following Cross-Cutting Area Components? Yes/No (Y/N) Document the basis for the safety culture Cross-Cutting Area Component as related or not.		Reference corrective actions addressing any safety culture Cross-Cutting Area Component noted as yes.
			Y/N	Basis	
Other	Organizational Change Management	CNP Management uses a systematic process for planning, coordinating, and evaluating the safety impacts of decisions related to major changes in organizational structures and functions, leadership policies, programs, procedures, and resources.	No	This evaluation found no crosscutting issues with organizational change management.	N/A
Other	Safety Policies	CNP's safety policies and related training establish and reinforce that nuclear safety is an overriding priority.	No	This evaluation found no crosscutting issues with CNP's safety policies.	N/A

Safety Culture Impact Review			
INPO 8 Principles of a Strong Nuclear Safety Culture			
Safety Culture: **Component Definition**	**Did the evaluation identify any of the following Component Definitions as issues? Yes/No (Y/N)** **Document the basis for the safety culture Cross- Component definition as related or not.**		**Reference corrective actions addressing any safety culture Cross-Cutting Area Component noted as yes.**
	Y/N	**Basis**	
Everyone is personally responsible for nuclear safety. At CNP, responsibility and authority for nuclear safety are defined and clearly understood. Reporting relationships, positional authority, staffing, and financial resources support nuclear safety responsibilities. Corporate policies emphasize the overriding importance of nuclear safety.	N	This evaluation found no crosscutting issues that affected nuclear safety culture.	N/A
Leaders demonstrate commitment to safety. At CNP, executive and senior managers are the leading advocates of nuclear safety and demonstrate their commitment both in word and action. The nuclear safety message is communicated frequently and consistently, occasionally as a stand-alone theme. Leaders throughout the nuclear organization set an example for safety.	N	This evaluation found no crosscutting issues that affected nuclear safety culture.	N/A
Trust permeates the organization. At CNP, a high level of trust is established in the organization, fostered, in part, through timely and accurate communication. There is a free flow of information in which issues are raised and addressed. Employees are informed of steps taken in response to their concerns.	N	This evaluation found no crosscutting issues that affected nuclear safety culture.	N/A

Safety Culture Impact Review			
INPO 8 Principles of a Strong Nuclear Safety Culture			
Safety Culture: **Component Definition**	**Did the evaluation identify any of the following Component Definitions as issues? Yes/No (Y/N)** **Document the basis for the safety culture Cross- Component definition as related or not.**		**Reference corrective actions addressing any safety culture Cross-Cutting Area Component noted as yes.**
	Y/N	**Basis**	
Decision-making reflects safety first. At CNP, personnel are systematic and rigorous in making decisions that support safe, reliable plant operation. Operators are vested with the authority and understand the expectation, when faced with unexpected or uncertain conditions, to place the plant in a safe condition. Senior leaders support and reinforce conservative decisions.	N	This evaluation found no crosscutting issues that affected nuclear safety culture.	N/A
Nuclear technology is recognized as special and unique. At CNP, the special characteristics of nuclear technology are taken into account in all decisions and actions. Reactivity control, continuity of core cooling, and integrity of fission product barriers are valued as essential, distinguishing attributes of the nuclear station work environment.	N	This evaluation found no crosscutting issues that affected nuclear safety culture.	N/A
A questioning attitude is cultivated. At CNP, individuals demonstrate a questioning attitude by challenging assumptions, investigating anomalies, and considering potential adverse consequences of planned actions. This attitude is shaped by an understanding that accidents often result from a series of decisions and actions that reflect flaws in the shared assumptions, values, and beliefs of the organization. All employees are watchful for conditions and activities that can have an undesirable effect on plant safety.	N	This evaluation found no crosscutting issues that affected nuclear safety culture.	N/A

Safety Culture Impact Review			
INPO 8 Principles of a Strong Nuclear Safety Culture			
Safety Culture: **Component Definition**	**Did the evaluation identify any of the following Component Definitions as issues? Yes/No (Y/N)** **Document the basis for the safety culture Cross- Component definition as related or not.**		**Reference corrective actions addressing any safety culture Cross-Cutting Area Component noted as yes.**
	Y/N	**Basis**	
Organizational learning is embraced. At CNP, OE is highly valued, and the capacity to learn from experience is well developed. Training, self-assessments, corrective actions, and benchmarking are used to stimulate learning and improve performance.	N	For this root cause evaluation, extensive searches of Operating Experience (OE) were performed. No OE was found that, if acted upon by CNP, would have prevented the issue. No issues were identified by this evaluation regarding self and independent assessments.	N/A
Nuclear safety undergoes constant examination. At CNP, oversight is used to strengthen safety and improve performance. Nuclear safety is kept under constant scrutiny through a variety of monitoring techniques, some of which provide an independent "fresh look."	N	This evaluation found no crosscutting issues that affected nuclear safety culture.	N/A

5.10 EQUIPMENT RELIABILITY REVIEW

Was the root or Apparent Cause of the equipment failure determined? Yes ☒ No ☐

- ☐ Equipment is verified as Run-to-failure and cause determination is not required. (Do not complete the rest of the form)
- ☐ ER process cause was not determined (External event, Historical, Unknown)(AP-913 PCND)

Comments:
N/A

1. Equipment Reliability Classification
Look up the criticality classification of the equipment involved in the Corrective Action Document.
Is the classification of the component correct? Yes ☐ No ☒

- ☒ Incorrect classification *(AP-913 INCORT)*
- ☐ Not Classified *(AP-913 NT_CLD)*

Comments/Corrective Actions:
The LRSS clevis bolts are piece parts, not a piece of equipment. Therefore, they are not classified as critical, non-critical, or run-to-failure. The Corrective Action Document lists the Reactor Vessel, 1-OME-1, as Critical, which is correct for the Reactor Vessel.

2. Performance Monitoring
Is the System Monitoring Plan and predictive maintenance performed on the equipment adequate?
Yes ☐ No ☐ NA ☒

- ☐ Monitored scope inadequate (e.g., levels, temp, pressures, Vibration) *(AP-913 PMSLTA)*
- ☐ Monitoring frequency not appropriate *(AP-913 PMFNA)*
- ☐ Monitoring execution less than adequate *(PMELTA)*
    - o Is the monitoring and threshold for action adequate?
    - o Is there improvement needed in collecting or trending the data?

Comments/Corrective Actions:
Performance monitoring of the LRSS clevis bolts is not performed because they are a structural component. However, inspection of the internals components is governed by the Reactor Vessel Internals Program Plan for Aging Management of Reactor Internals at D.C. Cook Nuclear Plant Unit 1 (WCAP-17300-NP).

3. Preventive Maintenance (PM)
Is PM program adequate? Yes ☐ No ☐ NA ☒

- ☐ PM did not exist *(AP-913 PMTDNE)*
- ☐ PM frequency not appropriate *(AP-913 PMFNA)*
- ☐ PM task content not appropriate (or less than adequate) *(AP-913 PMTCNA)*
- ☐ PM template/basis less than adequate *(AP-913 PMTTBL)*

☐ PM execution less than adequate *(AP-913 PMELTA)*
☐ PM feedback not implemented *(AP-913 PMTFNI)*

Comments/Corrective Actions:
A PM does not exist for the LRSS clevis bolts. However, inspection of the internals components is governed by the Reactor Vessel Internals Program Plan for Aging Management of Reactor Internals at D.C. Cook Nuclear Plant Unit 1 (WCAP-17300-NP).

4. Work Practices
Are the maintenance practices and behaviors appropriate and acceptable? Yes ☐ No ☐ NA ☒
☐ Work planning, instruction, or preparation less than adequate *(AP-913 WPIPLA)*
☐ PMT not performed or PMT less Than adequate *(AP-913 PMT)*
☐ Work activities incorrectly performed *(AP-913 WAIP)*

Comments/Corrective Actions:
Failures were not found to be the result of maintenance practices or behaviors. All possible causes in this category were refuted.

5. Design /Operation
Is the design of this component appropriate for the application? Yes ☐ No ☒
Are the operating procedures and practices appropriate? Yes ☒ No ☐ NA☐
☒ Original design less than adequate - Component not appropriate for its configuration/application *(AP-913 ORIG)*
☐ Design change less than adequate - Component not appropriate for its configuration/application *(AP-913 CHANGE)*
☐ Equipment was not operated within design *(AP-913 OPSNOWD)*

Comments/Corrective Actions:
The original design of the LRSS clevis bolts was determined to be less than adequate based on the bolts failing by PWSCC. The original design used an Alloy X-750 bolt in a heat treatment unknowingly susceptible to PWSCC. Additionally, all of the bolts failed in the head to shank transition due to the localized high stresses in this region of the bolt. The replacement bolts utilized an Alloy X-750 bolt with a less susceptible heat treatment and an improved head to shank design transition to reduce stress in this region.

6. Manufacturer/Vendor Quality, Procurement, Shipping, or Storage
Are parts availability and quality adequate? Yes ☐ No ☐ NA ☒
☐ Vendor quality or workmanship issues (manufacturing defects) *(AP-913 VQWI)*
☐ Procurement less than adequate (ex. Specification, Equivalence) *(AP-913 PLTA)*
☐ Receipt, Inspection, and Storage less than adequate (ex. Environment, Shelf Life, Control of Scavenged Parts, Storage PM) *(AP-913 RISLA)*

Comments/Corrective Actions:
LRSS clevis bolts installed at the Cook Nuclear Plant were original equipment manufacturer (OEM) components. Replacement bolts are not typically procured or stored.

7. Previous Corrective Action Implementation
Was corrective action to previous similar problems adequate? Yes ☐ No ☐ NA ☒
☐ Previous corrective actions less than adequate *or* untimely
☐ OE use less than adequate

Comments/Corrective Actions (if inadequate):
Degradation of LRSS clevis bolts had not been observed at Cook Nuclear Plant or elsewhere in the industry prior to this event.

8. Long Range Plan (Obsolescence/Life Cycle Management)
Is the Long Range Plan adequate? Yes ☒ No ☐ NA ☐
☐ Aging / obsolescence concern, Asset Management/LCM Plans less than adequate (AP-913 AOCAML)
☐ Previous Business Plan related items not implemented, untimely, or deferred *(AP-913 BPNIUD)*

Comments/Corrective Actions:
The Reactor Vessel Aging Management Program has been developed, but has not been completely implemented. This is a new program required for license renewal with a governing document that was completed in February 2011. Procedures must be updated and items must be put in the long range plan for inspections.

9. Other
Is configuration management complete and accurate for this CAP product? Yes ☐ No ☒ NA ☐
Is the equipment referenced in ActionWay correct? Yes ☒ No ☐ NA ☐

Comments/Corrective Actions:
N/A

10. Manufacture/Vendor Quality Check
Is there a concern with the quality of parts, shipping or handling? Yes ☐ No ☒ NA ☐

Comments/Corrective Actions:
Results of failure analysis indicate the failed bolts met the requirements of the OEM specification.

11. Problem/Issue Management Review
Have previous issues not been adequately addressed including but not limited to aging, obsolescence, chronic problem, scheduling, or business planning? Yes ☐ No ☐ NA ☒

12. Unknown or Different Cause
Did the equipment fail due to an unknown cause or other cause than listed in steps 1 through 11 above? Yes ☐ No ☒ NA ☐

5.11 APPLICABLE OE

Type	Doc.#	Title/Description	Review Comments
External	46 - OE 99 - 009452	In-Service Failure Of Alloy X-750 Cap Screw On Core Shroud Repair Tie Rod Assembly	During a refueling outage in 1999 at Nine Mile Point Unit 1, they found Alloy X-750 3/8" cap screws broken during a visual inspection of a core shroud repair tie rod assembly. Causes: IGSCC in conjunction with large, sustained differential thermal expansion stress due to fastening of dissimilar materials with the cap screw.
External	53 - OE19374	Failed RV Upper Internals Guide Tube Support Pins	During the Fall 2004 Farley Unit 1 reactor vessel upper internals guide tube support (split) pin replacement project, six (6) Alloy X-750 reactor vessel upper internals guide tube support pins failed due to Primary Water Stress Corrosion Cracking (PWSCC). Causes: Primary Water Stress Corrosion Cracking (PWSCC)
External	50 - SOER 84-5	Bolt Degradation or Failure in Nuclear Power Plants	Since 1974, an increasing number of bolt failures in reactor coolant systems have been reported at nuclear power plants. Many examples of Alloy X-750 material failure due to stress corrosion cracking.
External	51 - OE11453	Reactor Internal Split Pin Failures	During a Refueling outage on September 21, 2000, McGuire Unit 2, personnel discovered three cracked and broken guide tube support pins during an inspection. The inspection was the result of industry operating experience of similar support pin failures. Cause: Pure Water Stress Corrosion Cracking (PWSCC) after extended operation of Alloy X-750
External	54 - NUREG-1801_R2	Generic Aging Lessons Learned (GALL) Report	The Generic Aging Lessons Learned (GALL) Report from the NRC identifies the clevis insert bolts as having a possible aging effect/mechanism as loss of material due to wear. This report supports the use of EPRI MRP-227 for managing the aging effects of clevis insert bolts.
External	47 - NRC_IN82-29	Control Rod Drive (CRD) Guide Tube Support Pin Failures at Westinghouse	Since 1978, several failures of the control rod drive (CRD) Guide Tube Support Pins have occurred. The material is Alloy X-750, which, depending on the manufacturer and the fabrication date, has been solution heat-treated and age hardened at various temperatures and for various times.

Type	Doc.#	Title/Description	Review Comments
		PWRS	Cause: Stress Corrosion Cracking (SCC)
External	56 - OE25769	Five Control Rod Guide Tube Support Pins found failed during planned replacement activities	On October 15, 2007 during a refueling outage, Braidwood five (5) support pins were found to have failed. All five support pins came out of the Upper Internals in one piece; however, each had a leaf break off in the support pin removal station. Cause: Pressurized Water Stress Corrosion Cracking (PWSCC) of the Alloy X-750 pins.
External	63 - EAR PAR 06-047	Corrosion in Control Rod Guide Tube Support Pins	On 28-03-06, Vandellos II had an unplanned shutdown because of indications of loose parts detected possibly in the Steam Generator. Cause: Primary Water Stress Corrosion Cracking (PWSCC) and also to the phenomenon of Irradiation-Assisted Stress Corrosion Cracking (IASCC)
External	52 - OE13865	Unplanned Shutdown due to Loose Part in Steam Generator	On May 13, 2002, Wolf Creek had an unplanned Shutdown because of an indication that there was a loose part in their Steam Generator. After shutting down from 100% power a control rod guide tube split pin nut and locking device was found in the Steam Generator. Cause: Primary Water Stress Corrosion Cracking (PWSCC)
External	57 - OE7883	Shutdown because of detection of Foreign Object in Steam Generator	On May 29, 1996, Vogtle Unit 1 control room personal detect a loose part in the Steam Generator. This is the same scenario that Wolf Creek (above) went through in May of 2002. The part was also a nut from a control rod guide tube support pin. Another part was found lodged in the tube sheet and others were not located. Cause: Primary Water Stress Corrosion Cracking (PWSCC)
Internal	GT 2012-1808 / 2012-1809	D.C. Cook CRGT split pin GTLRP	The original material of the split pins was Alloy X-750 with a low temperature heat treat. These pins were susceptible to primary water stress corrosion cracking (PWSCC) and failed. Split pins were replaced by Areva in 1985 and 1986 with new Alloy X-750 split pins with improved stresses and a high temperature heat treat. These improved features reduced susceptibility to PWSCC, but did not

Type	Doc.#	Title/Description	Review Comments
			eliminate the problem. The comparison of the split pin failure to the clevis bolts was discussed by Westinghouse in WCAP-14577 Rev 1-A, March 2001. While it was known that the Alloy X-750 clevis bolt material was susceptible to PWSCC, the failure was not considered likely due to difference in fluence, temperature, and stresses between the two bolts.
Internal	AR 2010-10940	D.C. Cook Unit 1 core baffle bolt failures	This was the root cause evaluation for the core baffle bolt failures in the Unit 1 reactor vessel. The root cause of failed baffle-former bolts was Irradiation Assisted Stress Corrosion Cracking (IASCC) in conjunction with thermal and irradiation induced loss of preload in several baffle-former bolts. IASCC was considered in the Support/Refute analysis for this evaluation.

5.12 DOCUMENTS REVIEWED

All Documents are in the directory: \\cnp101\Engineering\Root Cause\LRSS Root Cause\LRSS Exhibits\

Documents Reviewed			
Type	**Exhibit - Document Name**	**Date Reviewed**	**Title/Description**
Report	1 - WCAP-12527 Connecticut Yankee Thermal Shield Removal pg225wordCaptureError.pdf Westinghouse Document WCAP-12527, Revision 0, *Connecticut Yankee Thermal Shield Removal Program Report.* April 1990.	11/14/13	Westinghouse – The purpose of this report is to address concerns of removing the Thermal Shield at Connectiut Yankee Nuclear Reactor
External OE	2 - OE2570 Thermal Shield Support Failure and Repair Connecticut Yankee.pdf INPO Document OE2570, *Thermal Shield Support Failure and Repair.* April 1988.	11/14/13	INPO - Thermal Shield Support Failure and Repair Connecticut Yankee
External OE	3 - Ringhals OE Lower Radial Support wear 2013.pdf Nilsson, P., *Internals Support Wear*. Presentation by Vattenfall to the Electric Power Research Institute. October 28,2013.	11/14/13	Vattenfall - Ringhals Lower Radial Support wear 2013
E-mail	4 - wilsonEmail20120924.pdf E-mail from Bryan Wilson, Westinghouse, to Kevin Kalchik, American Electric Power, *RE: MPR Review of LRSS Clevis Insert Bolting Analysis*. September 2012.	11/14/13	Westinghouse - Bryan Wilson's response to MPR Review of LRSS Clevis Insert Bolting Analysis
Report	5e - DC Cook Clevis Bolts FINAL Report 12-19-13.pdf Babcock & Wilcox Technical Services Group Report S-1473-002, *Examination of Clevis Bolts Removed from D. C. Cook Nuclear Plant.* December 2013.	11/14/13	B&W - This is an updated report covers laboratory examinations performed by Babcock & Wilcox Technical Services Group (B&W TSG) on failed clevis bolts removed from D. C. Cook Unit 1, Hot Cell Report. – Final Report

Documents Reviewed

Type	Exhibit - Document Name	Date Reviewed	Title/Description
Info Gram	6 - IG-10-1.pdf Westinghouse Document IG-10-1, Revision 0, *Reactor Internals Lower Radial Support Clevis Insert Cap Screw Degradation.* March 2010.	11/14/13	Westinghouse - Info Gram from Westinghouse of Cooks Reactor internals lower radial support clevis insert cap screw degradation
Video file	7 - Core Support Lugs.mpg CNP Document DC Cook U1C23 10 Year ISI, *Engineering Programs, DC Cook Outage U1C23 10 Year ISI Disk 1 of 4, WesDyne International – 1 DVD*, August 2010. CNP Document DC Cook U1C23 10 Year ISI, *Engineering Programs, DC Cook Outage U1C23 10 Year ISI Disk 2 of 4, WesDyne International – 1 DVD*, August 2010. CNP Document DC Cook U1C23 10 Year ISI, *Engineering Programs, DC Cook Outage U1C23 10 Year ISI Disk 3 of 4, WesDyne International – 1 DVD*, August 2010. CNP Document DC Cook U1C23 10 Year ISI, *Engineering Programs, DC Cook Outage U1C23 10 Year ISI Disk 4 of 4, WesDyne International – 1 DVD*, August 2010.	11/15/13	Video of U1C23 VT Inspection
Evaluation	8a - LTR-RCPL-10-41_R1.pdf Westinghouse Document LTR-RCPL-10-41, Revision 0, *Evaluations Supporting Operability Assessment for American Electric Power Service Corporation Donald C. Cook Nuclear Plant Unit 1.* March 2010.	11/14/13	Westinghouse - Evaluations Supporting Operability Assessment for American Electric Power Service Corporation (1 Cycle)

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Evaluation	8b - LTR-RCPL-10-41_R1.pdf Westinghouse Document LTR-RCPL-10-41, Revision 1, *Evaluations Supporting Operability Assessment for American Electric Power Service Corporation Donald C. Cook Nuclear Plant Unit 1.* January 2011.	11/14/13	Westinghouse - Evaluations Supporting Operability Assessment for American Electric Power Service Corporation (2 Cycles)
Analysis	9 - WCAP-17588-P Revision 1.pdf Westinghouse Document WCAP-17588-P, Revision 1, *D.C. Cook Unit 1 Lower Radial Support Clevis Insert Acceptable Minimum Bolting Pattern Analysis.* March 2013.	11/14/13	Westinghouse - Minimum Bolting Pattern Analysis
Evaluation	10 - LTR-RIDA-10-134[1].pdf Westinghouse Document LTR-RIDA-10-134, Revision 0, *D.C. Cook Unit 2 Engineering Evaluation of the Radial Support System Clevis insert Bolts and Operation through Spring 2012.* September 2010.	11/14/13	Westinghouse - D.C. Cook Unit 1 comparison to Unit 2 Engineering Evaluation of the Radial Support System Clevis Insert Bolts and Operation
Root Cause Investigation	11 - WCAP-15271 Rev. 0, Guide Tube Support Pin Degradation Root Cause Investigation.pdf Westinghouse Document WCAP-15271, Revision 0, *Guide Tube Support Pin Degradation Root Cause Investigation.* August 1999.	11/14/13	Westinghouse - Guide Tube Support Pin Degradation Root Cause Investigation from Westinghouse - Maanshan Unit 1
Technical Report	12 - matsHdbkNukeAppEPRI.pdf *Materials Handbook for Nuclear Plant Pressure Boundary Applications (2013).* EPRI, Palo Alto, CA: 2013. 3002000122.	11/14/13	EPRI - Materials Handbook for Nuclear Plant Pressure Boundary Applications
Presentation Slides	13a - MAI-Overview.pdf van der Lee, J., *The Materials Ageing Institute: General Overview.* The Materials Ageing Institute: Materials Degradation Course for Engineers in the Nuclear Industry, Vail, CO, USA, June 5-8, 2012.	11/14/13	The Materials Ageing Institute – General Overview

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Presentation Slides	13b - Nuclear Plant Environment Todd Allen.pdf Allen, T., *Nuclear Plant Environment.* The Materials Ageing Institute: Materials Degradation Course for Engineers in the Nuclear Industry, Vail, CO, USA, June 5-8, 2012.	11/14/13	The Materials Ageing Institute – Nuclear Plant Environment
Presentation Slides	13c - Operational Experience Peter Scott.pdf Scott, P., Vaillant, F., *Operational Experience – An Overview.* The Materials Ageing Institute: Materials Degradation Course for Engineers in the Nuclear Industry, Vail, CO, USA, June 5-8, 2012.	11/14/13	The Materials Ageing Institute – Operational Experience
Presentation Slides	13d - Materials and Their Use in Plant Components Francois Cattant.pdf Cattant, F., *Materials and Their Use in Plant Components.* The Materials Ageing Institute: Materials Degradation Course for Engineers in the Nuclear Industry, Vail, CO, USA, June 5-8, 2012.	11/14/13	The Materials Ageing Institute – Materials and Their Use in Plant Components Francois Cattant
Presentation Slides	13e - Fundamentals of Radiation Effects Allen .pdf Allen, T., *Fundamentals of Radiation Effects: What makes reactors special?.* The Materials Ageing Institute: Materials Degradation Course for Engineers in the Nuclear Industry, Vail, CO, USA, June 5-8, 2012.	11/14/13	The Materials Ageing Institute – Fundamentals of Radiation Effects
Presentation Slides	13f - PWR RPV Internals Anne Demma.pdf Demma, A., *Electric Power Research Institute: PWR Vessel Internals Integrity Issues.* The Materials Ageing Institute: Materials Degradation Course for Engineers in the Nuclear Industry, Vail, CO, USA, June 5-8, 2012.	11/14/13	EPRI – PWR RPV Internals

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Presentation Slides	13g - Fundamentals of Plant Chem Keith Fruzzetti.pdf Fruzzetti, K., *Electric Power Research Institute: Fundamentals of Plant Chemistry BWRs and PWRs.* The Materials Ageing Institute: Materials Degradation Course for Engineers in the Nuclear Industry, Vail, CO, USA, June 5-8, 2012.	11/14/13	EPRI – Fundamentals of Plant Chemistry
Report	14 - SCCinitiationStainlessInconelEPRI.pdf *Stress Corrosion Cracking Initiation Model for Stainless Steel and Nickel Alloys: Effects of Cold Work.* EPRI, Palo Alto, CA: 2009. 1019032.	11/14/13	EPRI - Stress Corrosion Cracking Initiation Model for Stainless Steel and Nickel Alloys, Effects of Cold Work
Chart	15-GDTChart.pdf *Geometric Dimensioning Chart*. Multimac, Athens, OH.	11/14/13	MulitMac - Geometric Dimensioning Chart
CAP Product	16 - AR 2010-1804.pdf CNP Document AR 2010-1804, *Rx Vessel Core Support Lug Bolting Anomalies*. Initiated March 21, 2010.	11/13/13	Rx Vessel Core Support Lug Bolting Anomalies
Correspondence	17 - AEP-11-5.pdf Westinghouse Document AEP-11-5, Revision 0, *American Electric Power Donald C. Cook Unit 1 Engineering Services for One (1) Cycle Extension of Unit 1 Operability Determination*. February 2011.	11/14/13	Westinghouse - Correspondence to Phil Lozmack from Daniel Beddingfield extending repair plan for one Cycle.
Evaluation	18 - EVAL-10-21.pdf Westinghouse Document EIES-10-67, Revision 0, *Revision to D.C. Cook-1 Loose Parts Operability Determination (LTR-RCPL-10-41) and 50359 Screen (EVAL-10-21)*. December 2010.	11/14/13	Westinghouse - Revision to D.C. Cook-1 Loose Parts Operability Determination (LTR-RCPL-10-41) and 50.59 Screen (EVAL-10-21)

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Type	Exhibit - Document Name	Date Reviewed	Title/Description
Excel Document	19 - boltInspectionResultsSummary.xlsx CNP Document, *19-boltInspectionReultsSummary.xlsx*. Microsoft Excel file, April 2013.	11/14/13	Side by Side picture comparison of Clevis Bolts and Dowel Pins from U1C23 to U1C25
Evaluation	20 - CN-RIDA-10-27_R2.pdf Westinghouse Document CN-RIDA-10-27, Revision 2, *Structural Evaluation of As-Found Degraded Bolting Pattern for D.C. Cook Unit 1 Lower Radial Support Clevis Insert*. January 2011.	11/14/13	Westinghouse - Structural Evaluation of As-Found Degraded Bolting Pattern for D.C. Cook Unit 1 Lower Radial Support Clevis Insert
Evaluation	21 - LTR-RIDA-10-75_R0.pdf Westinghouse Document LTR-RIDA-10-75, Revision 0, *Evaluation of Potential for Loose Parts due to Damaged Core Barrel Support Lug Bolts for D.C. Cook Unit 1*. March 2010.	11/14/13	Westinghouse- Evaluation of Potential for Loose Parts due to Damaged Core Barrel Support Lug Bolts for D.C. Cook Unit 1
Evaluation	22 - LTR-RIDA-10-76_R1.pdf Westinghouse Document LTR-RIDA-10-76, Revision 1, *D.C. Cook Unit 1 Lower Radial Support Clevis Cap Screw Locking Bar Wear Evaluation*. November 2010.	11/14/13	Westinghouse - D.C. Cook Unit 1 Lower Radial Support Clevis Cap Screw Locking Bar Wear Evaluation
Evaluation	23 - LTR-RIDA-10-78_R2.pdf Westinghouse Document LTR-RIDA-10-78, Revision 2, *Structural Evaluation Summary of As-Found Degraded Bolting Pattern for D.C. Cook Unit 1 Lower Radial Support Clevis Insert*. January 2011.	11/14/13	Westinghouse - Structural Evaluation Summary of As-Found Degraded Bolting Pattern for D.C. Cook Unit 1 Lower Radial Support Clevis Insert
Evaluation	24 - LTR-RIDA-10-82_R1.pdf Westinghouse Document LTR-RIDA-10-82, Revision 1, *D.C. Cook Clevis Insert Bolts: Stress Corrosion Cracking of Inconel X-750*. December 2010.	11/14/13	Westinghouse - D.C. Cook Clevis Insert Bolts: Stress Corrosion Cracking of Alloy X-750

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Type	Exhibit - Document Name	Date Reviewed	Title/Description
Evaluation	25 - LTR-RIDA-10-337.pdf Westinghouse Document LTR-RIDA-10-337, Revision 0, *Evaluation of Commercial Risks Associated with Continued Operation with Degraded Clevis Insert Bolts at D.C. Cook Unit 1*. January 2011.	11/14/13	Westinghouse - Evaluation of Commercial Risks Associated with Continued Operation with Degraded Clevis Insert Bolts at D.C. Cook Unit 1
Evaluation	26 - PE-10-18_R1.pdf Westinghouse Document PE-10-18, Revision 1, *D.C. Cook Unit 1 Lower Radial Support Clevis Cap Screw Locking Bar Wear Evaluation – Product Engineering Input, Revision 1*. November 2010.	11/14/13	Westinghouse - D.C. Cook Unit 1 Lower Radial Support Clevis Cap Screw Locking Bar Wear Evaluation – Product Engineering Input, Revision 1
Drawing	27 - AEP On-site As Built Loop 4 Reactor Internals Drawing 108D467.pdf Westinghouse Drawing 108D467, Revision 2, *AEP As-Built 4 Loop Reactor Internals On-Site As-Built*. December 1983.	11/18/13	Westinghouse - On-site As Built Loop 4 Reactor Internals
Drawing	28 - 1 in Socket Headed Cap Screw Drawing 206C037.pdf Westinghouse Drawing 206C037, Revision 2, *1.000 Soc. Hd. Cap Screw (Undercut)*. March 1969.	11/18/13	Westinghouse - 1 in Socket Headed Cap Screw
Drawing	29 - PWR Lower Radial Support Insert Customizing Drawing 541F473.pdf Westinghouse Drawing 541F473, Revision 4, *173-00-000-000 PWR Lower Radial Support Insert Customizing*. April 1972.	11/18/13	Westinghouse - PWR Lower Radial Support Insert Customizing

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Drawing	30 - PWR Lower Radial Support Insert Preliminary Drawing 541F620.pdf Westinghouse Drawing 541F620, Revision 1, *173-00-000-000 PWR Lower Radial Support Insert Preliminary Machining*. August 1968.	11/18/13	Westinghouse - PWR Lower Radial Support Insert Preliminary
Drawing	31 - Lower Radial Support Clevis Insert Gaging and Assembly 685J790-S1-R4.PDF Westinghouse Drawing 685J790, Sheet 1, Revision 4, *173-00-000-000 PWR Lower Radial Support Clevis Insert Gaging & Assembly*. November 1971.	11/18/13	Westinghouse - Lower Radial Support Clevis Insert Gaging and Assembly
Drawing	32 - Core Barrel Support Lug Drawing Sheet 2 685J790-S2-R4.PDF Westinghouse Drawing 685J790, Sheet 2, Revision 4, *173-00-000-000 PWR Lower Radial Support Clevis Insert Gaging & Assembly*. November 1971.	11/18/13	Westinghouse - Core Barrel Support Lug Sheet 2
Drawing	33 - Core Barrel Support Lug Drawing Sheet 3 685J790-S3-R4.PDF Westinghouse Drawing 685J790, Sheet 3, Revision 4, *173-00-000-000 PWR Lower Radial Support Clevis Insert Gaging & Assembly*. November 1971.	11/18/13	Westinghouse - Core Barrel Support Lug Sheet 3
Drawing	34 - Core Barrel Support Lug Drawing Sheet 4 685J790-S4-R4.PDF Westinghouse Drawing 685J790, Sheet 4, Revision 4, *173-00-000-000 PWR Lower Radial Support Clevis Insert Gaging & Assembly*. November 1971.	11/18/13	Westinghouse - Core Barrel Support Lug Sheet 4

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Drawing	35 - AEP Reactor (Internals) as Built Drawing Fig 12 and 14 5656D81_6.pdf Westinghouse Drawing 5656D81, Sheet 7, Revision 2, *AEP Reactor (Internals) As Built Drawings Fig 13 & 14*. May 1973.	11/18/13	Westinghouse - Reactor (Internals) as Built Figure 12 and 14
Drawing	36 - AEP Reactor as Built Drawing Fig 15 5656D81_8.pdf Westinghouse Drawing 5656D81, Sheet 8, Revision 2, *AEP Reactor (Internals) As Built Drawings Fig 13 & 14*. May 1973.	11/18/13	Westinghouse - Reactor as Built Figure 15
Drawing	37 - AEP Reactor Vessel as Built Drawing Fig 9 and 10 DC-132486.pdf Westinghouse Drawing 108D002, Sheet 5, Revision 2, *AEP Reactor Vessel As Builts Fig 9 & 10*. May 1971.	11/18/13	Westinghouse - Reactor Vessel as Built Figure 9 and 10
Drawing	38 - General Arrangement - Elevation Drawing E-233-440.pdf Combustion Engineering Drawing 233-440, Revision 2, *General Arrangement – Elevation For Westinghouse Electric Corp. 173" I.D. Reactor Vessel*. March 1969.	11/18/13	Westinghouse - General Reactor Arrangement - Elevation
Drawing	39 - Bottom Head Forming and Welding Drawing E-233-443.pdf Combustion Engineering Drawing 233-443, Revision 2, *Bottom Head Forming & Welding For Westinghouse Electric Corp. 173" I.D. Reactor Vessel*. December 1969.	11/18/13	Westinghouse - Bottom Head Forming and Welding
Drawing	40 - Core Support Lug Drawing E-233-450_2.tif Combustion Engineering Drawing 233-450, Revision 2, *Miscellaneous Attachments for Westinghouse Electric Corp. 173" I.D. Reactor Vessel*. September 1968.	11/18/13	Westinghouse - Core Support Lug

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Specifications	41 - Inconel_alloy_X-750.pdf Special Metals Publication No. SMC-067, *Inconel alloy X-750*. Special Metals, Huntington, WV. September 2004.	11/18/13	Special Metals - Specs from Special Metals on Inconel Alloy X750
Report	42-EPRI NP-7338-L.pdf *Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175)*. EPRI, Palo Alto, CA: 2005. 1012081.	11/19/13	EPRI - Design and Manufacturing Guidelines for High-Strength Components in LWRs-Alloy X-750
Journal Article	43 - journalArticleX750.pdf Mills, J.J., Lebo, M.R., and Kearns, J.J., *Hydrogen Embrittlement, Graind Boundary Segregation, and Stress Corrosion Cracking of Alloy X-750 in Low- and High- Temperature Water*, Metallurgical and Materials Transactions A, Volume 30A, June 1999, pp. 1579-1596.	11/18/13	Metallurgical and Materials Transactions Volume 30A - Hydrogen Embrittlement, Grain Boundary Segregation, and Stress Corrosion Cracking of Alloy X-750 in Low and High-Temperature Water
Report	44 - EPRI Design and Manufacturing Guidelines for Alloy X750.pdf *Design and Manufacturing Guidelines for High-Strength Components in LWRs – Alloy X-750*. EPRI, Palo Alto, CA: 1991. NP-7338-L.	11/19/13	EPRI - Design and Manufacturing Guidelines for Alloy X-750
Report	45 - NP-6392-SD Microstructure and SCC resistance of X-750_718_286.pdf *Microstructure and Stress Corrosion Resistance of Alloys X-750, 718, and A-286 in LWR Environments*. EPRI, Palo Alto, CA: 1989. NP-6392-SD.	11/19/13	EPRI - Microstructure and Stress Corrosion Resistance of Alloy X-750, 718, and A-286 in LWR Environments

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External OE	46 - OE 99 - 009452.pdf INPO Document OE10154, *In-Service Failure Of Alloy X-750 Cap Screw On Core Shroud Repair Tie Rod Assembly – rod assembly upper spring assembly*. August 1999.	11/20/13	INPO - OE from Nine Mile Point Unit 1
NRC Information Notice	47 - NRC_IN82-29.pdf NRC Information Notice IN 82-29, *Control Rod Drive (CRD) Guide Tube Support Pin Failures at Westinghouse PWRs*. July 1982.	11/20/13	NRC - OE from NRC Information Notice No. 82-29
Report	48 - RVIAMP_R0a_FINAL.pdf CNP Document *Reactor Vessel Internals Aging Management Program*. September 2012.	11/20/13	AEP - Donald C. Cook Nuclear Plant Reactor Vessel Internals Aging Management Program
Presentation Slides	49 - clevis_insert_bolts_dec2010.ppt Lott, R., *Clevis Insert Bolt Issue Update*. Pressurized Water Reactor Owners Group: Materials Subcommittee. Marco Island, FL: December 7-9, 2010.	11/20/13	PWR Owners Group - Clevis Insert Bolt Issue Update - Randy Lott
External OE	50 - SOER 84-5.pdf INPO Document SOER 84-5, *Bolt Degradation or Failure in Nuclear Power Plants*. September 1984.	11/20/13	INPO - Bolt Degradation or Failure in Nuclear Power Plants
External OE	51 - OE11453.pdf INPO Document OE11453, *Reactor Internal Split Pin Failures*. October 2000.	11/20/13	INPO - OE from McGuire Unit 2
External OE	52 - OE13865.pdf INPO Document OE13865, *Unplanned Shutdown due to Loose Part in Steam Generator*. August 2004.	11/20/13	INPO - OE from Wolf Creek

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External OE	53 - OE19374.pdf INPO Document OE19374, *Failed RV upper Internals Guide Tube Support Pins*. October 2004.	11/20/13	INPO - OE from Farley Unit 1
Report	54 - 13627 -1801_R2.pdf NRC Document NUREG-1801, Revision 2, *Generic Aging Lessons Learned (GALL) Report*. December 2010.	11/20/13	NRC - Generic Aging Lessons Learned (GALL) Report (rev. 2)
Report	55 - WCAP-14577R1-A.pdf Westinghouse Document WCAP-14577, Revision 1-A, *License Renewal Evaluation: Aging Management for Reactor Internals*. March 2001.	11/20/13	Westinghouse - License Renewal Evaluation: Aging Management for Reactor Internals
External OE	56 - OE25769.pdf INPO Document OE25769, *Five Control Rod Guide Tube Support Pins Found Failed During Planned Replacement Activities*. November 2007.	11/20/13	INPO - OE from Braidwood Unit 1
External OE	57 - OE7883.pdf INPO Document OE7883, *Shutdown Because of Detection of Foreign Object in Steam Generator*. October 1998.	11/20/13	INPO - OE from Vogtle Unit 1
Evaluation	58 - LTR-RIDA-10-134[1].pdf Westinghouse Document LTR-RIDA-10-134, Revision 0, *D.C. Cook Unit 2 Engineering Evaluation of the Radial Support System Clevis insert Bolts and Operation through Spring 2012*. September 2010. [Redundant to Exhibit 10]	11/20/13	Westinghouse - D.C. Cook Unit 2 Engineering Evaluation of the Radial Support System Clevis Insert Bolts and Operation through Spring 2012

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Type	**Exhibit - Document Name**	**Date Reviewed**	**Title/Description**
Time line	59 - Timeline for Unit 1 Reactor Pressure Vessel internals examinations.docx CNP Document, *Unit 1 RPV 10 Year Visual Examinations*. 59 - Timeline for Unit 1 Reactor Pressure Vessel internals examinations.docx. Microsoft Word File, November 2013.	11/20/13	Roy Hall - Timeline for Unit 1 Reactor Pressure Vessel internals examinations
Time line	60 - Timeline for Unit 2 Reactor Pressure Vessel internals examinations.docx CNP Document, *Unit 2 RPV 10 Year Visual Examinations*. 60 - Timeline for Unit 2 Reactor Pressure Vessel internals examinations.docx. Microsoft Word File, November 2013.	11/21/13	Roy Hall - Timeline for Unit 2 Reactor Pressure Vessel internals examinations
Information Slides	61 - CI_Designs.pdf Westinghouse Slides, *Geometry Comparisons*. Provided November 2013.	11/21/13	Westinghouse - Geometry Comparisons of Clevis Insert Designs
Internal OE	62 - OE30993.pdf INPO Document OE30993, *During 10-Year Reactor Vessel In-Service Inspection Seven Failed Core Barrel Lower Lateral Restraint Bolts were Observed*. March 2010.	11/22/13	INPO - OE of DC Cook Clevis bolting degradation
External OE	63 - EAR PAR 06-047.pdf EAR PAR 06-047, *Corrosion in Control Rod Guide Tube Support Pins*. March 2006.	11/22/13	INPO - OE of Vandellos II
Periodical	64 - nuclearNewsPlantListing2011ANS.pdf Nuclear News, Volume 54, Number 3, *13th Annual Reference Issue*. American Nuclear Society, Inc., LaGrange Park, IL: March 2011.	11/25/13	Nuclear News – 13th Annual Reference Issue

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Design Information Transmittal	65 - DIT-S-00705-17.pdf CNP Document DIT-S-00705-17, *Unit 1 and Unit 2 Burnup Data.* October 2013.	11/25/13	AEP – Donald C. Cook Nuclear Plant Units 1 & 2 - Burnup Data
Program Document	66 - ISI Program.pdf CNP Document DCC01.G03, Revision 3, *Donald C. Cook Nuclear Plant Units 1 & 2 ISI Program Plan Fourth Ten-Year Inspection Interval.* January 2013.	11/25/13	AEP - Donald C. Cook Nuclear Plant Units 1 & 2 - ISI Program Plan Fourth Ten-Year Inspection Interval
Memorandum	67 - U1thermalSheidIndication.pdf CNP Memo From D. A. Patience to R. L. Otte, *Unit 1 Thermal Shield Indication*. September 10, 1985.	11/25/13	AEP - Resolution of the surface indication found on the Unit 1 thermal shield on June 30
Report	68 - WCAP-17484-P.pdf Westinghouse Document WCAP-17484-P, Revision 0, *Reactor Internals Aging Management MRP-227-A Implementation Manual for Westinghouse – designed Nuclear Steam Supply Systems*. August 2012.	11/26/13	Westinghouse - Reactor Internals Aging Management MRP-227-A Implementation Manual for Westinghouse designed Nuclear Steam Supply Systems
Report	69 - WCAP-14522_RCA_U1_Barrel_Bolt.pdf Westinghouse Document WCAP-14522, Revision 0, *Root Cause Determination of Core Barrel-Baffle Former Bolting Failure at Donald C. Cook Nuclear Plant, Unit 1*. December 1995.	11/26/13	Westinghouse - Root Cause Determination of Core Barrel - Baffle Former Bolting Failure at DC Cook Nuclear Plant, Unit 1
Working Notes	70 - seatSurfAcceptCrit20130425.doc LRSS Engineering Working Notes, *LRSS Bolt Seating Surface Acceptance Criteria*. April 2013.	11/26/13	AEP - This document provides requirements of clevis seating surfaces for replacement clevis bolts.

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Design Change	71 - 1-DCP-0125.pdf CNP Document 1-DCP-0125, Revision 0, *Reactor Vessel Core Barrel*. June 1997.	11/26/13	AEP - Design Change Package - 01-0125, Replace barrel-former bolts at locations A-4, A-5, and A-6
CAP Product	72 - GT2013-1241.pdf CNP Document GT 2013-1241, *IER Lvl 3 (13-2) – Dislodged Thermal Sleeves*. Initiated January 28, 2013.	11/26/13	AEP - GT 2013-1241, Dislodged Thermal Sleeves
Video Files	73 - VIDEO_TShighlights CNP Unit 1 Video Highlights from Barrel-Former Bolt Discovery and Repairs, 1994-1997.	11/26/13	AEP - Folder of Videos from the Inspection and repair of the barrel-former Bolts
Report	74 - MRP-191.pdf *Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191)*. EPRI, Palo Alto, CA: 2006. 1013234.	11/26/13	EPRI - Materials Reliability Program - This report describes the process and results of categorizing Westinghouse and Combustion engineering designed pressurized water reactor (PWR) internals components according to age-related degradation and significance.
Report	75 - WCAP-17300-NP Cook AMP Unit 1.pdf Westinghouse Document WCAP-17300-NP, Revision 0, *Reactor Vessel Internals Program Plan for Aging Management of Reactor Internals at D.C. Cook Nuclear Plant Unit 1*. February 2011.	11/26/13	Westinghouse - Reactor Vessel Internals Program Plan for Aging Management of Reactor Internals at D.C. Cook Nuclear Plant Unit 1
Quality Assurance Data Package	76a - 23-9202929-000, QADP special 1 inch bolts.pdf AREVA Document 23-9202929, Revision 0, *Quality Assurance Data Package: LRSS Clevis Replacement Contingency Bolt Assembly Standard Threads For AEP / DC Cook Unit 1*. April 2013.	11/26/13	Areva - LRSS Clevis Replacement Contingency Bolt Assembly Standard Threads For AEP / DC Cook Unit 1 AEP Contract 1500256 Release Number 55 AREVA NP Inc. Contract No. F.501639

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Quality Assurance Data Package	76b - 23-9195126-001 QADP for 5 remachined bolts.pdf AREVA Document 23-9195126, Revision 1, *Quality Assurance Data Package: LRSS Clevis Replacement Bolt Assemblies For AEP / DC Cook Unit 1*. April 2013.	11/26/13	Areva - LRSS Clevis Replacement Bolt Assemblies For AEP / DC Cook Unit 1 AEP PO 01559035 AREVA NP Inc. Contract No. F.501639
Quality Assurance Data Package	76c - 23-9195126-002.pdf AREVA Document 23-9195126, Revision 2, *Quality Assurance Data Package: LRSS Clevis Replacement Bolt Assemblies For AEP / DC Cook Unit 1*. April 2013.	11/26/13	Areva - LRSS Clevis Replacement Bolt Assemblies For AEP / DC Cook Unit 1 AEP PO 01559035 Rev. 001 & 002 AREVA NP Inc. Contract No. F.501639
Instant Message Record	77 - Roy E Hall_AEPIN-20131210-1042.html Instant Message Conversation between Kevin Kalchik, American Electric Power, and Roy Hall, American Electric Power. December 10, 2013.	12/10/13	Instant message conversation between Kevin Kalchik and Roy Hall discussing ASME Code inspections, code relief, and core barrel pulls
CAP Product	78 - AR 00124109 Missile Block Drop.pdf CNP Document AR 00124109, *A stop work order has been initiated in response to a*. Initiated March 27, 2006	11/26/13	AEP - AR written documenting Missile Block drop in Unit 2
Report	79 - MRP-80.pdf *Materials Reliability Program: A Review of Thermal Aging Embrittlement in Pressurized Water Reactors (MRP-80)*, EPRI, Palo Alto, CA: 2003.1003523.	12/2/13	EPRI - Materials Reliability Program - This report documents the results of a review of PWR materials summarizing the available data and recommendations for follow-on testing to determine the effects of thermal aging.
Presentation Slides	80 - W Cook Clevis April 2010 MSC.pdf Lott, R., *Radial Support Clevis Bolting Fractures: Industry Implications*. Pressurized Water Reactor Owners Group: Materials Subcommittee. Lake Buena Vista, FL: April 20-22, 2010.	12/3/13	Westinghouse – Plant Ranking Based on PWSCC Assumption

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Procedure	81 - 12-OHP-4050-FHP-044.pdf CNP Document 12-OHP-4050-FHP-044, Revision 10, *Reactor Vessel Lower Internals Removal*. January 2013.	12/3/13	AEP - Reactor Vessel Lower Internals Removal
Procedure	82 - 12-OHP-4050-FHP-045.pdf CNP Document 12-OHP-4050-FHP-045, Revision 11, *Reactor Vessel Lower Internals Replacement*. January 2013.	12/3/13	AEP - Reactor Vessel Lower Internals Replacement
Report	83 - LTR-RIDA-13-183_6.pdf Westinghouse Document LTR-RIDA-13-183, Revision 0, *Material Specifications for Lower Radial Support System (LRSS) Clevis Insert Bolts*. December 2013.	12/4/13	Westinghouse – The purpose of this letter is to provide the applicable material specifications for the lower radial support system (LRSS) clevis insert bolts at D. C. Cook Unit 1
Report	84 - MRP-134.pdf *Materials Reliability Program: Framework and Strategies for Managing Aging Effects in PWR Internals (MRP-134)*. EPRI, Palo Alto, CA: 2005. 1008203.	12/03/13	EPRI - Materials Reliability Program – This report describes a framework and associated strategies for managing effects of aging in PWR internals.
Report	85a - MRP-156.pdf *Materials Reliability Program: Pressurized Water Reactor Issue Management Table, PWR-IMT Consequence of Failure (MRP-156)*. EPRI, Palo Alto, CA: 2005. 1012110.	12/03/13	EPRI - Materials Reliability Program – This report provides initial input to the Issue Management Table to address the consequences of failure for the identified components in a reactor coolant system.
Report	85b - MRP-156supplement.pdf EPRI Supplement, *Attention Recipientts of EPRI Report 1012110*. EPRI, Palo Alto, CA: December 2005.	12/03/13	EPRI - Materials Reliability Program – This Report is to document some potential errors and omissions that were discovered during reviews of MRP-156.

Documents Reviewed

Type	Exhibit - Document Name	Date Reviewed	Title/Description
Report	86 - MRP-191.pdf *Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191)*. EPRI, Palo Alto, CA: 2006. 1013234.	12/03/13	EPRI - Materials Reliability Program – This report describes the process and results of categorizing Westinghouse and Combustion Engineering designed PWR internals components according to age-related degradation and significance.
Report	87a - MRP-227.pdf *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0)*. EPRi, Palo Alto, CA: 2008. 1016596.	12/03/13	EPRI - Materials Reliability Program – PWR Internals inspection and evaluation Guidelines (rev. 0)
Report	87b - MRP-227-A.pdf *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)*. EPRI, Palo Alto, CA: 2011. 1022863.	12/03/13	EPRI - Materials Reliability Program - PWR Internals inspection and evaluation Guidelines (A)
Report	88 - MRP-232.pdf *Materials Reliability Program: Aging Management Strategies for Wesitnghouse and Combustion Engineering PWR Internals (MRP-232)*. EPRI, Palo Alto, CA: 2008. 1016593.	12/03/13	EPRI - Materials Reliability Program – This report provides the technical basis for the aging management requirements of Westinghouse and Combustion Engineering reactor internals
Report	89 - NUREG-1801 Vol 2 Chapter XI Section M.pdf NRC Document NUREG-1801, Revision 0, *Generic Aging Lessons Learned (GALL) Report, Chapter XI: Aging Management programs (AMPs)*. April 2001.	12/04/13	NRC - Generic Aging Lessons Learned (GALL) Report (Chapter XI)
Spreadsheet	90 - coreBurnupData.xlsx CNP Document, *90 - coreBurnupData.xlsx*. Microsoft Excel file, December 2013.	12/09/13	AEP - Shows Effective Degradation Years for Unit 1 and Unit 2

Documents Reviewed			
Type	**Exhibit - Document Name**	**Date Reviewed**	**Title/Description**
Report	91 - WCAP-13627.pdf Westinghouse Document WCAP-13627, Revision 0, *Pilot Application of Risk-Based Inspection Methods to Westinghouse-Designed Reactor internals*. June 1993.	12/04/13	Westinghouse - This report provides the methods and results of a Westinghouse Owners Group pilot study to demonstrate and set precedence for the application of risk-based technologies in the development of nuclear power plant component inspection programs.
Procedure	92 - 12-EHP-5034-SPV-001.pdf CNP Document 12-EHP-5034-SPV-001, Revision 0, *Single point Vulnerability Management*. March 2013.	12/04/13	AEP - Single Point Vulnerability Management
Report	93 - ap-913 rev 4.pdf INPO Document AP-913, Revision 4, *Equipment Reliability Process Description*. October 2013.	12/04/13	INPO – This document describes an equipment reliability process offered to assist member utilities to maintain high levels of safe and reliable plant operation in an efficient manner
Presentation Slide	94 - dowel_pin_loosening_scenario.pdf Westinghouse Slides, *Broken Dowel Pin Tack Weld and Pin Loosening Scenario*. Provided December 2013.	12/05/13	Westinghouse – Broken Dowel Pin Tack Weld and Pin Loosening Scenario
Procedure	95 - DTG-EQR-002.pdf CNP Document DTG-EQR-002, Revision 5, *Component Scoping*, September 2012.	12/04/13	AEP - Component Scoping
Procedure	96 - EHI-5054-RPV.pdf CNP Document EHI-504-RPV, Revision 1, *Reactor Vessel Integrity*, May 2013.	12/04/13	AEP - Reactor Vessel Integrity

Documents Reviewed			
Type	**Exhibit - Document Name**	**Date Reviewed**	**Title/Description**
Report	97 - LRP-EAMP-01.pdf CNP Document LRP-EAMP-01, Revision 3, *License Renewal Project: Evaluation of Aging Management Programs*. November 2005.	12/04/13	Areva - License Renewal Project Evaluation of Aging Management Programs (rev. 3)
Report	98 - LRP-MAMR-01.pdf CNP Document LRP-MAMR-01, Revision 3, *Aging Management Review of the Reactor Coolant System*. September 2005.	12/04/13	AEP - License Renewal Aging Management Review of the RCS (rev. 3)
Report	99 - WCAP-16198-P- Alloy 600 Program Cook.pdf Westinghouse Document WCAP-16198-P, Revision 1, *PWSCC Susceptibility Assessment of the Alloy 600 and Alloy 82/182 Components in D. C. Cook Units 1 and 2*. July 2004.	12/05/13	Westinghouse - PWSCC Susceptibility Assessment of the Alloy 600 and Alloy 82/182 Components in D.C. Cook Units 1 and 2
Report	100 - WCAP-15271 Rev. 0, Guide Tube Support Pin Degradation Root Cause Investigation.pdf Westinghouse Document WCAP-15271, Revision 0, *Guide Tube Support Pin Degradation Root Cause Investigation*. August 1999.	12/05/13	Westinghouse – Guide Tube Support Pin Degradation Root Cause Investigation
Procedure	101 - EHI-5070-ALLOY600.pdf CNP Document EHI-5070-ALLOY600, Revision 4, *Alloy 600 Material Management Program*. January 2012.	12/05/13	AEP - Alloy 600 Material Management Program
Letter	102 - ML13325A973.pdf Florida Power & Light Company letter L-2013-287 to the USNRC, dated October 30, 2013, *License Renewal (LR) Reactor Vessel Internals (RVI) Inspection Program Response to Request for Additional Information (RAI)*, Agencywide Documents and Accession Management System (ADAMS) Accession No. ML13325A973.	12/05/13	Florida Power & Light - License Renewal Reactor Vessel Internals Inspection Program Response to RAIs

Documents Reviewed			
Type	**Exhibit - Document Name**	**Date Reviewed**	**Title/Description**
Letter	103 - ML13270A250.pdf Entergy Nuclear Northeast letter NL-13-122 to the USNRC, dated September 27, 2013, *Reply to Request for Additional Information Regarding the License Renewal Application Indian Point Nuclear Generating Units Nos. 2 & 3*, Agencywide Documents and Accession Management System (ADAMS) Accession No. ML13270A250.	12/05/13	Entergy – IPEC License Renewal
Report	104 - WCAP-16777-NP.pdf Westinghouse Document WCAP-16777, Revision 0, *Interim Report on Aging Management Strategies for Westinghouse and Combustion Engineering Reactor Vessel Internals Components*, April 2007.	12/05/13	Westinghouse – Management Strategies for Westinghouse and Combustion Engineering Reactor Vessel Internals Components
Accepted Stores Procedure Package	105 - Flow Restrictor ASP Package.pdf CNP Document ASP-27585, Revision 0, *Westinghouse – Covers Restrictor Flow Assy*. April 2003.	12/05/13	AEP - Unit 1 and 2 Flow Restrictor ASP Package
–Design Information Transmittal	106 - Flow Restrictor DIT.pdf CNP Document DIT-B-03240-00, *D.C. Cook unit 2 Part Length Control Rod Guide Tube Flow Restrictors*. October 2007.	12/05/13	AEP - D.C. Cook Unit 2 Part Length Control Rod Guide Tube Flow Restrictors
Report	107 - WCAP-11000 U2 failed CRGT pin op rpt.pdf Westinghouse Document WCAP-11000, Revision 0, *D. C. Cook Unit 2 Estimated Operability with Failed Control Rod Gide Tube Support Pins*. January 1985.	12/05/13	Westinghouse – Estimated Operability with Failed Control Rod Guide Tube Support Pins
Procedure	108 - U1 Spring 2010 Lower Internals Removal Procedure.pdf CNP Document 12-OHP-4050-FHP-044, Revision 8, *Reactor Vessel Lower Internals Removal*. March 2010.	12/06/13	AEP- Procedure used during U1C23 removal of the Internals under WO 55293911-07

Documents Reviewed			
Type	**Exhibit - Document Name**	**Date Reviewed**	**Title/Description**
Data Graph	109 - Unit 2 Tcold.pdf CNP Data Retrieved From R*Time Points: *PPC2.DATA.T0406A RCS LP 1 COLD LEG TEMPER (SNAP), PPC2. DATA.T0426A RCS LP 2 COLD LEG TEMPER (SNAP), PPC2.DATA.T0446A RCS LP 3 COLD LEG TEMPER (SNAP), PPC2.DATA.T0466A RCS LP 4 COLD LEG TEMPER (SNAP)*. 1/1/2003 to 12/9/2013, Snap Data Period 24 hrs. Retrieved December 9, 2013.	12/09/13	AEP – Graph of Cold Leg Temps of Unit 2 January 2003 to December 2013
Data Graph	110 - Unit 1 Tcold.pdf CNP Data Retrieved from R*Time Points: *PPC1.DATA.T0406A RCS LOOP 1 COLD LEG TEMP (SNAP), PPC1.DATA.T0426A RCS LOOP 2 COLD LEG TEMP (SNAP), PPC1.DATA.T0446A RCS LOOP 3 COLD LEG TEMP (SNAP), PPC1.DATA.T0466A RCS LOOP 4 COLD LEG TEMP (SNAP)*. 1/1/2003 to 12/9/2013, Snap Data Period 24 hrs. Retrieved December 9, 2013.	12/09/13	AEP – Graph of Cold Leg Temps of Unit 1 January 2003 to December 2013
Data Graph	111 - Unit 1 Tcold Turbine Outage.pdf CNP Data Retrieved from R*Time Points: *PPC1.DATA.T0406A RCS LOOP 1 COLD LEG TEMP (SNAP), PPC1.DATA.T0426A RCS LOOP 2 COLD LEG TEMP (SNAP), PPC1.DATA.T0446A RCS LOOP 3 COLD LEG TEMP (SNAP), PPC1.DATA.T0466A RCS LOOP 4 COLD LEG TEMP (SNAP)*. 9/20/2008 to 11/22/2009, Snap Data Period 24 hrs. Retrieved December 9, 2013.	12/09/13	AEP – Graph of Cold Leg Temps of Unit 1 During Turbine Shutdown
–Design Information Transmittal	112 - DIT-B-03415-00.pdf CNP Document DIT-B-03415-00, *Justification for Continued Operation for U2C19 Regarding LRSS*. August 2010.	12/10/13	AEP – Justification for Continued Operation for U2C19 Regarding LRSS

Documents Reviewed			
Type	**Exhibit - Document Name**	**Date Reviewed**	**Title/Description**
Report	113 - MRP-205.pdf *Materials Reliability Program: Pressurized Water Reactor issue Management Tables – Revision 3 (MRP-205)*. EPRI, Palo Alto, CA: 2013. 3002000634.	12/10/13	EPRI - Materials Reliability Program – This PWR IMT report identifies, describes, and prioritizes 80 open R&D gaps where additional research is needed to resolve issues related to degradation of PWR NSS components.
Report	114 - LTR-RIDA-13-188.pdf Westinghouse Document LTR-RIDA-13-188, Revision 0, *D.C. Cook units 1 and 2 Reactor Internals Assembly Specification – 616A225 Rev. 4*. December 2013.	12/11/13	Westinghouse – The purpose of this letter is to provide the reactor internals assembly specification for D.C. Cook Unit 1 and Unit 2.
E-Mail	115 - e-mail on clevis lugs.pdf E-mail from Kevin Neubert, Westinghouse, to Kevin Kalchik, American Electric Power, *RE: Responses to Cook RCA ActionItemList20131127.xlsx*. December 11, 2013.	12/11/13	Westinghouse – Responses to Cook RCA Action Items List
E-Mail	116 - thermal shield flexure material.pdf E-mail from Kevin Neubert, Westinghouse, to Kevin Kalchik, American Electric Power, *Cook Units 1 and 2 – Thermal Shield Flexure*. December 11, 2013.	12/11/13	Westinghouse – Cook Unit 1 and Unit 2 Thermal Shield Flexure
Purchase Requisition	117 - 1988 U2 RPV Exam.pdf CNP Document 01681-040-8X, *ISI Reactor Vessel Examinations*. March 1988.	12/12/13	AEP – Purchase Requisition for the mechanized ultrasonic examinations and remote visual examination of Unit 2 Reactor Vessel for the 1988 ISI inspection
Licensee Event Report	118 - chloride LER 1.pdf CNP Licensee Event Report 82-071/03L-0 Transmittal Letter with Report Attached From W.G. Smith, American Electric Power, to J.G. Keppler, USNRC. September 7, 1982.	12/11/13	AEP - Licensee Event Report from 1982

Documents Reviewed			
Type	**Exhibit - Document Name**	**Date Reviewed**	**Title/Description**
Licensee Event Report	119 - chloride LER.pdf CNP Licensee Event Report 81-009/03L-0 Transmittal Letter with Report Attached From D.V. Shaller, American Electric Power, to J.G. Keppler, USNRC. April 29, 1981.	12/11/13	AEP - Licensee Event Report from 1981
Procedure Figure	120 - cooldown hydrogen control band.pdf CNP Document 12-THP-6020-CHM-110 – Figure 1, Revision 36, *RCS Chemistry – Shutdown and Refueling: RCS Cooldown Hydrogen Control Band*. October, 2013.	12/12/13	AEP - 12-THP-6020-CHM-110 Figure 1 RCs Cool Down Dissolved Hydrogen
Report	121 - MRP-274.pdf *Materials Reliability Program: Assessment of Westinghouse and Combustion Engineering Nickel-Based Alloy Orphan Locations (MRP-274)*. EPRI, Palo Alto, CA: 2010. 1021025.	12/12/13	EPRI - This report evaluates the potential for aging degradation of components inside the reactor vessel that are fabricated from austenitic nickel-based alloys and their associated weld metals and that have not been addressed in detail in other documents and programs.
Report	122 - MRP-48.pdf *PWR Materials Reliability Program Response to NRC Bulletin 2001-01 (MRP-48)*, EPRI, Palo Alto, CA: 2001. 1006284.	12/12/13	EPRI - This report contains the plant rankings using the time-at-temperature model and provides comments regarding applicable regulatory requirements.
Inspection Plan	123 - DC Cook Unit-2 RPV Visual Inspection Plan (2009) (Rev-3) (3).pdf AREVA Document, *DC Cook Unit-2, Outage 2009 U2C-18 RPV Visual Examinations*. AREVA Contract No. A001717. March 2009.	12/13/13	AREVA-RPV Visual Examination ISI Inspection Plan for U2C18

Documents Reviewed

Type	Exhibit - Document Name	Date Reviewed	Title/Description
Report	124 - Underwater Remote Visual Examination of the Reactor Vessel and Internals Unit 1 Cycle 23 RFO.pdf CNP Document DC Cook U1C23 10 Year ISI, *Engineering Programs, DC Cook Outage U1C23 10 Year ISI Disk 1 of 4, WesDyne International – 1 DVD*, August 2010. CNP Document DC Cook U1C23 10 Year ISI, *Engineering Programs, DC Cook Outage U1C23 10 Year ISI Disk 2 of 4, WesDyne International – 1 DVD*, August 2010. CNP Document DC Cook U1C23 10 Year ISI, *Engineering Programs, DC Cook Outage U1C23 10 Year ISI Disk 3 of 4, WesDyne International – 1 DVD*, August 2010. CNP Document DC Cook U1C23 10 Year ISI, *Engineering Programs, DC Cook Outage U1C23 10 Year ISI Disk 4 of 4, WesDyne International – 1 DVD*, August 2010.	12/13/13	Underwater remote visual examination of the Reactor Vessel and Internals of U1C23 Summary Report
Procedure	125 - 12-THP-6020-CHM-110 (rev. 8).pdf CNP Document 12-THP-6020-CHM-110, Revision 8, *RCS Chemistry-Shutdown/Refueling*. January 2002.	12/13/13	AEP – 12-THP-6020-CHM-110 rev. 8
Spreadsheet Data and Graph	126 - Turbine Outage Hydrogen.xls CPN Data Retrieved from WinCDMS Database Point: *Unit 1-RCS-H_2*. 9/1/2008 to 12/31/2010. Retrieved December 13, 2013.	12/13/13	AEP – Graph of Hydrogen during the U1 Turbine Event
Report	127 - MPR-118.pdf *Materials Reliability Program: Suitability of Emerging Technologies for Mitigation of PWSCC (MRP-118)*. EPRI, Palo Alto, CA: 2004. 1009500.	12/13/13	EPRI – The goals of this report were to review new technologies and to determine if they were viable for PWR applications for mitigating PWSCC.

Documents Reviewed			
Type	**Exhibit - Document Name**	**Date Reviewed**	**Title/Description**
CAP Product	128 - GT 2013-18674-1 CARB Approval of Mid Brief.pdf CNP Document GT 2013-18674-1, *CARB 828 Meeting Minutes Tracking*. Initiated December 6, 2013.	12/13/13	AEP - Mid CARB Brief on Reactor Vessel Core Support Lug Bolting Anomalies
Meeting Agenda	129 - CARB 828 Agenda with LRSS Mid Brief Material.pdf CNP Document, *Management Screening Committee (MSC) / Corrective Action Review Board (CARB) #828 Agenda*. December 6, 2013.	12/13/13	AEP - Mid CARB Brief on Reactor Vessel Core Support Lug Bolting Anomalies
Report	130 - DZO White PaperZinc 2010.doc Miller, D.W., *Management White Paper Technical Evaluation of Zinc Addition For Cook, Units 1 & 2*. December 28, 2010.	12/16/13	AEP - Technical Evaluation of Zinc Addition For Cook, Units 1 & 2
Report	131a - LTR-RIDA-13-189.pdf Westinghouse Document LTR-RIDA-13-189, Revision 0, *D.C. Cook Unit 1 Lower Radial Support System Quality Control System Requirements, Trip Reports and Field Deficiencies*. December 2013.	12/18/13	Westinghouse - D.C. Cook Unit 1 Lower Radial Support System Quality Control Systems Requirements, Trip Reports and Field Deficiencies
Specification	131b - QCS-1_R3.pdf_1387291934_LTR-RIDA-13-189.pdf Westinghouse Document QCS-1, Revision 3, *Manufacturer's Quality Control System Requirements*. May 1967.	12/18/13	Westinghouse - This specification establishes requirements for manufacturer's systems for control of quality during manufacture, including inspection plans.
Trip Reports	131c - AEP_Field_Deficienties_and_Trip_Reports.pdf_1387293684_LTR-RIDA-13-189.pdf Westinghouse Document LTR-RIDA-13-189, Attachments 2 through 5, Revision 0, *D.C. Cook Unit 1 Lower Radial Support System Quality Control Systems Requirements, Trip Reports and Field Deficiencies*. December 2013.	12/18/13	Westinghouse - Surveillance covering the installation of the clevis inserts and reinstallation of the lower internals assembly for the final measurement of the clevis fit and nozzle gaps.

Documents Reviewed			
Type	**Exhibit - Document Name**	**Date Reviewed**	**Title/Description**
Report	132 - MRP-280.pdf *Materials Reliability Program: Mitigation of Stress Corrosion Crack Growth in Nickel-Based Alloys in Primary Water by Hydrogen Optimization and Zinc Addition (MRP-280).* EPRI, Palo Alto, CA: 2010. 1021013.	12/18/13	EPRI- The objective of this report was to determine effective methods involving changes in primary water chemistry for the mitigation of PWSCC in thick-wall components constructed of Alloy 600 and its weld metals in PWR coolant systems.
Regulatory Guide	133 - RG1.133_ML003740137.pdf USNRC Regulatory Guide 1.133, Revision 1, *Loose-Part Detection Program for the primary System of Light-Water-Cooled Reactors.* May 1981. ML003740137.	2/24/14	Loose parts monitoring Reg Guide
Report	134 - RPT-0025-1304-1 - MPR Oversight for B&W Hot Cell Analysis and Testing of Failed LRSS Clevis Bolts.pdf MPR Document RPT-0025-1304-1, Revision 0, *Summary MPR Technical Oversight of B&W Hot Cell Laboratory Analysis and Testing of Failed LRSS Clevis Bolts from D.C. Cook Unit 1.* February 2014.	2/25/14	Letter report documenting MPR oversight of B&W hot cell testing and report
Vendor Technical Document	135 - VTD-FANP-0001.pdf CNP Document VTD-FANP-0001, Revision 1, *Framatome ANP, Inc. Instruction manual for Loose Parts Monitoring System – (LPMS-V) [PUB. #01-5021870-00].* August 2008.	2/25/14	CNP loose parts monitoring instruction manual

5.13 CARB COMMENTS

CARB Comments:

Page 4, first paragraph under Basis, remove last sentence starting with "It is not..."

- DONE

Page 7, section 1.6.3, remove paragraph starting with "PMP-7030_CAP-005..." and the associated action.

- DONE

Page 7, section 1.6.3, define the due dates on the actions.

- DONE

Page 34, add discussion concerning the means to detect loose parts.

- DONE, added RG 1.133 and VTD-FANP-0001 to references to support discussion

NRC Resident Administrative Comments:

Clarify reason for bolt replacement when there is not safety or operability concern.

- DONE, Added discussion starting on Page 25.

Annotate proprietary information because the NRC may be subject to public request for information, this will protect proprietary information.

- DONE, no proprietary information found in the LRSS RCE
- Hot Cell Report will be marked proprietary before submitting to the NRC

Include more discussion in section 3.3 indicating that CNP was aware of X-750 issues and followed industry guidance.

- DONE, indicated that CNP follows ASME Code, EPRI-MRP guidance, and PWROG-MSC guidance

Define "Literature" in supporting information column for 1.4 Corrosion Failure in the Potential Failure mode Evidence Support/Refute Matrix.

- DONE, referenced EPRI Materials Handbook

Provide more detail in the Reference section such that a reader outside CNP may be able to find and retrieve references.

- DONE, references section updated with more explicit detail

Administrative CARB Comments:

No administrative comments received from any CARB member